Facility: Grand Gulf Nuclear Station Date of Exam: May 2017 RO Outline

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	2	3	4	N/A			3	4	N/A		4	20			
	2	1	1	1				1	1			2	7			
	Tier Totals	3	4	5				4	5			6	27			
2. Plant Systems	1	2	2	3	3	2	2	2	3	2	2	3	26			
	2	1	1	1	1	1	1	1	1	1	1	2	12			
	Tier Totals	3	3	4	4	3	3	3	4	3	3	5	38			
3. Generic Knowledge and Abilities Categories					1		2		3		4		10	1 / 2	3 / 4	
					2		4		2		2					

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only 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)								Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A2	G*	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4						X	G2.1.7 Ability to evaluate plant performance and make operational judgements based on operating characteristics reactor behavior, and instrument interpretation. (CFR: 41.5)	4.4	1
295003 Partial or Complete Loss of AC / 6					X		Ability to determine and/or interpret the following as they apply to a PARTIAL OR COMPLETE LOSS OF A.C. POWER; AA2.01: Cause of partial or complete loss of A.C. Power (CFR 41.1)	3.4	2
295004 Partial or Total Loss of DC Pwr / 6		X					Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: AK2.01: Battery Charger (CFR 41.7)	3.1	3
295005 Main Turbine Generator Trip / 3				X			Ability to operate and/or monitor the following as they apply to MAIN TURBINE TRIP: AA1.02: RPS (CFR: 41.7)	3.6	4
295006 SCRAM / 1			X				Knowledge of the reasons for the following responses as they apply to SCRAM : AK3.04: Reactor water level setpoint setdown (CFR: 41.5)	3.1	5
295016 Control Room Abandonment / 7							Not sampled		
295018 Partial or Total 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 : AK1.01 Effects on component/system operations (CFR: 41.8 to 41.10)	3.5	6
295019 Partial or Total Loss of Inst. Air / 8			X				Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : AK3.02: Standby air compressor operation (CFR: 41.5)	3.5	7
295021 Loss of Shutdown Cooling / 4				X			Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING : AA1.05 Reactor recirculation. (CFR: 41.7 / 45.6)	3.0	8
295023 Refueling Acc / 8		X					Knowledge of the interrelations between REFUELING ACCIDENTS and the following: AK2.02: Fuel pool cooling and cleanup system. (CFR: 41.7)	2.9	9

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO)		Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A2	G*	K/A Topic(s)	IR	#
295024 High Drywell Pressure / 5					X		Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: EA2.10: (CFR: 41.10)	3.7	10
295025 High Reactor Pressure / 3						X	G2.2.39; Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10)	3.9	11
295026 Suppression Pool High Water Temp. / 5						X	G2.2.42: Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10)	3.9	12
295027 High Containment Temperature / 5	X						Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY): EK1.02: Reactor water level measurement: Mark-III (CFR: 41.8 to 41.10)	3.0	13
295028 High Drywell Temperature / 5			X				Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : EK3.04: Increased drywell cooling (CFR: 41.5)	3.6	14
295030 Low Suppression Pool Wtr Lvl / 5				X			Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: EA1.04: Suppression pool make-up system: Mark-III (CFR: 41.7	4.0	15
295031 Reactor Low Water Level / 2		X					Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: EK2.13: ARI/RPT/ATWS: Plant-Specific (CFR: 41.7)	4.1	16
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1					X		Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : EA2.06: Reactor pressure (CFR: 41.10)	4.0	17
295038 High Off-site Release Rate / 9						X	G2.1.27, Knowledge of system purpose and/or function. (CFR: 41.7)	3.9	18
600000 Plant Fire On Site / 8			X				Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: AK3.04: Actions contained in the abnormal procedure for plant fire on site	2.8	19

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO)		Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A2	G*	K/A Topic(s)	IR	#
700000 Generator Voltage and Electric Grid Disturbances / 6					X		Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AA2.04: Voltage outside the generator capability curve (CFR: 41.5)	3.6	20
K/A Category Totals:	2	3	4	3	4	4	Group Point Total:		20

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3							Not sampled		
295007 High Reactor Pressure / 3	X						Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : AK1.01 - Pump shutoff head (CFR: 41.8 to 41.10)	2.9	21
295008 High Reactor Water Level / 2						X	2.1.20 Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12)	4.6	22
295009 Low Reactor Water Level / 2							Not sampled		
295010 High Drywell Pressure / 5							Not sampled		
295011 High Containment Temp / 5							Not sampled		
295012 High Drywell Temperature / 5							Not sampled		
295013 High Suppression Pool Temp. / 5							Not sampled		
295014 Inadvertent Reactivity Addition / 1							Not sampled		
295015 Incomplete SCRAM / 1					X		AA2. Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM: (CFR: 41.10 / 43.5 / 45.13) AA2.01 Reactor power	4.1	23
295017 High Off-site Release Rate / 9							Not sampled		
295020 Inadvertent Cont. Isolation / 5 & 7			X				AK3. Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: AK3.01 Reactor SCRAM (CFR: 41.5 / 45.6)	3.8	24
295022 Loss of CRD Pumps / 1		X					AK2. Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: AK2.03 Accumulator pressures (CFR: 41.7 / 45.8)	3.4	25
295029 High Suppression Pool Wtr Lvl / 5						X	2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. (CFR: 41.7 / 45.7 / 45.8)	4.5	26
295032 High Secondary Containment Area Temperature / 5							Not sampled		

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO)						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	#
295033 High Secondary Containment Area Radiation Levels / 9							Not sampled		
295034 Secondary Containment Ventilation High Radiation / 9				X			Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : EA1.02 Process radiation monitoring system (CFR: 41.7 / 45.6)	3.9	27
295035 Secondary Containment High Differential Pressure / 5							Not sampled		
295036 Secondary Containment High Sump/Area Water Level / 5							Not sampled		
500000 High CTMT Hydrogen Conc. / 5							Not sampled		
K/A Category Point Totals:	1	1	1	1	1	2	Group Point Total:		7

ES-401 BWR Examination Outline Form ES-401-1

Plant Systems - Tier 2/Group 1 (RO)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode		X						X				Knowledge of electrical power supplies to the following: K2.01: Valves (CFR: 41.7) 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: A2.09: Inadequate system flow (CFR: 41.5)	3.5 3.3	28 29
205000 Shutdown Cooling			X									Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: K3.04 Recirculation loop temperatures (CFR: 41.7 / 45.4)	3.7	30
206000 HPCI												N/A for GGNS		
20700 Isol Condenser												N/A for GGNS		
209001 LPCS						X						K6. Knowledge of the effect that a loss or malfunction of the following will have on the LOW PRESSURE CORE SPRAY SYSTEM: K6.01 A.C. power (CFR: 41.7 / 45.7)	3.4	31
209002 HPCS					X							K5. Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): K5.04 Adequate core cooling: BWR-5,6 (CFR: 41.5 / 45.3)	3.8	32
211000 SLC											X	2.4.6 Knowledge of EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)	3.7	33

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
212000 RPS	X						X					K1. Knowledge of the physical connections and/or cause effect relationships between REACTOR PROTECTION SYSTEM and the following: K1.03 Recirculation system (CFR: 41.2 to 41.9 / 45.7 to 45.8) A1. Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: A1.01 RPS motor-generator output voltage (CFR: 41.5 / 45.5)	3.4 2.8	34 35
215003 IRM					X							K5. Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : K5.03 Changing detector position (CFR: 41.5 / 45.3)	3.0	36
215004 Source Range Monitor				X								K4. Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: K4.01 Rod withdrawal blocks (CFR: 41.7)	3.7	37
215005 APRM / LPRM			X									K3. Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: K3.01 RPS (CFR: 41.7 / 45.4)	4.0	38
217000 RCIC				X					X			K4. Knowledge of REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) design feature(s) and/or interlocks which provide for the following: K4.02 Prevent over filling reactor vessel (CFR: 41.7) A3. Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: A3.04 System flow (CFR: 41.7 / 45.7)	3.3 3.6	39 40

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
218000 ADS		X										K2. Knowledge of electrical power supplies to the following: K2.01 ADS logic (CFR: 41.7)	3.1	41
223002 PCIS/Nuclear Steam Supply Shutoff								X				A2. 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: A2.04 Process radiation monitoring system failures (CFR: 41.5 / 45.6)	2.9	42
239002 SRVs									X			A3. Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including: A3.01 SRV operation after ADS actuation (CFR: 41.7 / 45.7)	3.8	43
259002 Reactor Water Level Control										X	X	A4. Ability to manually operate and/or monitor in the control room: A4.02 All individual component controllers in the automatic mode (CFR: 41.7 / 45.5 to 45.8) G2.1.32 Ability to explain and apply system limits and precautions. (CFR: 41.10 / 43.2 / 45.12)	3.7 3.8	44 45
261000 SGTS						X						K6 Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : K6.01 A.C. electrical distribution (CFR: 41.7 / 45.7)	2.9	46
262001 AC Electrical Distribution							X					A1. Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including: A1.03 Bus voltage (CFR: 41.5 / 45.5)	2.9	47

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
262002 UPS (AC/DC)	X											K1. Knowledge of the physical connections and/or cause effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: K1.14 Main steam line radiation monitors (CFR: 41.2 to 41.9 / 45.7 to 45.8)	2.8	48
263000 DC Electrical Distribution			X									K3. Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: K3.03 Systems with D.C. components (CFR: 41.7 / 45.4)	3.4	49
264000 EDGs								X			X	A2. 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: A2.03 Operating unloaded, lightly loaded, and highly loaded (CFR: 41.5 / 45.6) 2.1.30 Ability to locate and operate components, including local controls. (CFR: 41.7 / 45.7)	3.4 4.4	50 51
300000 Instrument Air										X		A4. Ability to manually operate and / or monitor in the control room: A4.01 Pressure gauges (CFR: 41.7 / 45.5 to 45.8)	2.6	52
400000 Component Cooling Water				X								K4. Knowledge of CCWS design feature(s) and or interlocks which provide for the following: K4.01 Automatic start of standby pump (CFR: 41.7)	3.4	53
K/A Category Point Totals:	2	2	3	3	2	2	2	3	2	2	3	Group Point Total:		26

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
201001 CRD Hydraulic							X					A1. Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM controls including: A1.03 CRD system flow (CFR: 41.5 / 45.5)	2.9	54
201002 RMCS												N/A for GGNS		
201003 Control Rod and Drive Mechanism					X							K5. Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM : K5.08 How control rods affect shutdown margin (CFR: 41.5 / 45.3)	3.1	55
201004 RSCS												N/A for GGNS		
201005 RCIS			X									K3. Knowledge of the effect that a loss or malfunction of the ROD CONTROL AND INFORMATION SYSTEM (RCIS) will have on following: K3.02 Reactor startup: BWR-6 (CFR: 41.7 / 45.4)	3.5	56
201006 RWM												N/A for GGNS		
202001 Recirculation								X				A2. Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.08 Recirculation flow mismatch: Plant-Specific (CFR: 41.5 / 45.6)	3.1	57
202002 Recirculation Flow Control		X										K2. Knowledge of electrical power supplies to the following: K2.02 Hydraulic power unit: Plant-Specific (CFR: 41.7)	2.6	58
204000 RWCU									X			A3. Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including: A3.03 Response to system isolations (CFR: 41.7 / 45.7)	3.6	59
214000 RPIS												N/A for GGNS		
215001 Traversing In-Core Probe												Not sampled		

ES-401 BWR Examination Outline Form ES-401-1

Plant Systems - Tier 2/Group 2 (RO)

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
215002 RBM												N/A for GGNS		
216000 Nuclear Boiler Inst.											X	2.1.20 Ability to interpret and execute procedure steps (CFR: 41.10 / 43.5 / 45.12)	4.6	60
219000 RHR/LPCI: Torus/Pool Cooling Mode												Not sampled		
223001 Primary CTMT and Aux.	X											K1. Knowledge of the physical connections and/or cause effect relationships between PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES and the following: K1.14 RCIC: Plant-Specific (CFR: 41.2 to 41.9 / 45.7 to 45.8)	3.3	61
226001 RHR/LPCI: CTMT Spray Mode				X								K4. Knowledge of RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE design feature(s) and/or interlocks which provide for the following: K4.01 Testability of all operable components (CFR: 41.7)	2.6	62
230000 RHR/LPCI: Torus/Pool Spray Mode												N/A for GGNS		
233000 Fuel Pool Cooling/Cleanup												Not sampled		
234000 Fuel Handling Equipment												Not sampled		
239001 Main and Reheat Steam										X		A4. Ability to manually operate and/or monitor in the control room: A4.01 MSIV's . (CFR: 41.7 / 45.5 to 45.8)	4.2	63
239003 MSIV Leakage Control												Not sampled		
241000 Reactor/Turbine Pressure Regulator												Not sampled		
245000 Main Turbine Gen. / Aux.												Not sampled		
256000 Reactor Condensate												Not sampled		
259001 Reactor Feedwater						X						K6. Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR FEEDWATER SYSTEM : K6.02 Condensate system (CFR: 41.7 / 45.7)	3.3	64
268000 Radwaste												Not sampled		
271000 Offgas												Not sampled		
272000 Radiation Monitoring												Not sampled		

ES-401											BWR Examination Outline Plant Systems - Tier 2/Group 2 (RO)	Form ES-401-1	
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
286000 Fire Protection											X	2.1.30 Ability to locate and operate components, including local controls.	4.4	65
288000 Plant Ventilation												Not sampled		
290001 Secondary CTMT												Not sampled		
290003 Control Room HVAC												Not sampled		
290002 Reactor Vessel Internals												Not sampled		
204000 RWCU												Not sampled		
K/A Category Point Totals:	1	1	1	1	1	1	1	1	1	1	2	Group Point Total:		12

Facility: Grand Gulf Nuclear Station		Date of Exam: May 2017				
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.4	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc. CFR 41.10	3.3	66		
	2.1.36	Knowledge of procedures and limitations involved in core alterations. CFR: 41.10	3.0	67		
	Subtotal			2		
2. Equipment Control	2.2.1	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity. CFR: 41.5	4.5	68		
	2.2.13	Knowledge of tagging and clearance procedures. CFR: 41.10	4.1	69		
	2.2.21	Knowledge of pre- and post-maintenance operability requirements. CFR: 41.10	2.9	70		
	2.2.35	Ability to determine Technical Specification Mode of Operation. CFR: 41.7	3.6	71		
	Subtotal			4		
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions. CFR: 41.12	3.2	72		
	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	3.2	73		
	Subtotal			2		
4. Emergency Procedures / Plan	2.4.3	Ability to identify post-accident instrumentation. CFR: 41.6	3.7	74		
	2.4.20	Knowledge of the operational implications of EOP warnings, cautions, and notes. | CFR: 41.10	3.8	75		
	Subtotal			2		
Tier 3 Point Total				10		

Facility: Grand Gulf Nuclear Station Date of Exam: May 2017 SRO Outline

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1													3	4	7
	2													2	1	3
	Tier Totals													5	5	10
2. Plant Systems	1													2	3	5
	2													1 1	1	3
	Tier Totals													4	4	8

3. Generic Knowledge and Abilities Categories	1	2	3	4		1	2	3	4	7
						2	1	2	2	

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

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A2	G*	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4									
295003 Partial or Complete Loss of AC / 6					X		Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: AA2.02: Reactor power/pressure/and level (CFR: 43.5)	4.3	76
295004 Partial or Total Loss of DC Pwr / 6									
295005 Main Turbine Generator Trip / 3									
295006 SCRAM / 1									
295016 Control Room Abandonment / 7									
295018 Partial or Total Loss of CCW / 8									
295019 Partial or Total Loss of Inst. Air / 8									
295021 Loss of Shutdown Cooling / 4									
295023 Refueling Acc / 8						X	G2.4.4: Ability to recognize abnormal indications for system operating parameters that are entry level conditions for emergency and abnormal procedures (CFR 43.2)	4.7	77
295024 High Drywell Pressure / 5					X		Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: EA2.09: Containment Pressure MK-III (CFR: 43.5)	4.1	78
295025 High Reactor Pressure / 3					X		Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE EA2.02: Reactor Power (CFR: 43.5)	4.2	79
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						X	G2.4.6: Knowledge of EOP mitigation strategies (CFR 43.5)	4.7	80
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1									

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (SRO)		Form ES-401-1
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A2	G*	K/A Topic(s)	IR	#
295038 High Off-site Release Rate / 9						X	G2.4.21: Knowledge of 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 43.5)	4.6	81
600000 Plant Fire On Site / 8									
700000 Generator Voltage and Electric Grid Disturbances / 6						X	G2.2.22: Knowledge of LCOs and safety limits (CFR 43.2)	4.7	82
K/A Category Totals:	0	0	0	0	3	4	Group Point Total:		7

ES-401 BWR Examination Outline Form ES-401-1

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (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									
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2									
295010 High Drywell Pressure / 5									
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5						X	2.2.40 Ability to apply Technical Specifications for a system. (CFR: 41.10 / 43.2 / 43.5 / 45.3)	4.7	83
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1									
295017 High Off-site Release Rate / 9					X		Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: AA2.01: Offsite release rate (CFR 43.5)	4.2	84
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									
295034 Secondary Containment Ventilation High Radiation / 9					X		Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: EA2.02: Causes of high rad levels (CFR 43.5)	4.2	85

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (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	#
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:	0	0	0	0	2	1	Group Point Total:		3

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode														
205000 Shutdown Cooling														
206000 HPCI												N/A for GGNS		
20700 Isol Condenser												N/A for GGNS		
209001 LPCS														
209002 HPCS											X	G2.1.23: Ability to perform specific system and integrated plant procedures during all modes of plant operations (CFR 45.2)	4.4	86
211000 SLC								X				Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions: A2.04: Inadequate flow (CFR 41.5)	3.4	87
212000 RPS														
215003 IRM														
215004 Source Range Monitor														
215005 APRM / LPRM														
217000 RCIC								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: A2.15: Steam line break (CFR 41.5)	3.8	88
218000 ADS														
223002 PCIS/Nuclear Steam Supply Shutoff											X	G2.2.12 Knowledge of Surveillance procedures (CFR 43.2)	4.1	89
239002 SRVs														
259002 Reactor Water Level Control														
261000 SGTS														

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
262001 AC Electrical Distribution														
262002 UPS (AC/DC)														
263000 DC Electrical Distribution														
264000 EDGs											X	G2.2.25 Knowledge of Bases in the Tech Specs for LCOs and safety limits (CFR 43.2)	4.2	90
300000 Instrument Air														
400000 Component Cooling Water														
K/A Category Point Totals:	0	0	0	0	0	0	0	2	0	0	3	Group Point Total:		5

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
201001 CRD Hydraulic														
201002 RMCS												N/A for GGNS		
201003 Control Rod and Drive Mechanism														
201004 RSCS												N/A for GGNS		
201005 RCIS								X				Ability to (a) predict the impacts of the following on the ROD CONTROL AND INFORMATION SYSTEM (RCIS); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.13: Rod drift BWR-6 (CFR 41.5)	3.8	92
201006 RWM												N/A for GGNS		
202001 Recirculation														
202002 Recirculation Flow Control														
204000 RWCU														
214000 RPIS												N/A for GGNS		
215001 Traversing In-Core Probe														
215002 RBM												N/A for GGNS		
216000 Nuclear Boiler Inst.														
219000 RHR/LPCI: Torus/Pool Cooling Mode														
223001 Primary CTMT and Aux.														
226001 RHR/LPCI: CTMT Spray Mode														
230000 RHR/LPCI: Torus/Pool Spray Mode												N/A for GGNS		
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment			X									Knowledge of the effect that a loss or malfunction of the FUEL HANDLING EQUIPMENT will have on the following: K3.04: core modifications/alterations (CFR 41.7)	3.8	93
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator														
245000 Main Turbine Gen. / Aux.														
256000 Reactor Condensate														

ES-401								BWR Examination Outline Plant Systems - Tier 2/Group 2 (SRO)						Form ES-401-1
System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
259001 Reactor Feedwater														
268000 Radwaste														
271000 Offgas														
272000 Radiation Monitoring														
286000 Fire Protection														
288000 Plant Ventilation														
290001 Secondary CTMT														
290003 Control Room HVAC											X	G2.2.38 Knowledge of conditions and limits in the facility license (CFR 43.2)	4.5	91
290002 Reactor Vessel Internals														
204000 RWCU														
K/A Category Point Totals:	0	0	1	0	0	0	0	1	0	0	1	Group Point Total:		3

Facility: Grand Gulf Nuclear Station		Date of Exam: May 2017 SRO Outline				
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.5	Ability to use procedures related to shift staffing, such as minimum crew compliment, overtime limits, etc (CFR 43.5)			3.9	94
	2.1.34	Knowledge of primary and secondary plant chemistry limits (CFR 43.5)			3.5	95
	Subtotal					2
2. Equipment Control	2.2.5	Knowledge of the process for making design or operating changes to the facility (CFR 43.3)			3.2	96
	Subtotal					1
3. Radiation Control	2.3.6	Ability to approve release permits (CFR 43.4)			3.8	97
	2.3.13	Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to rad monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high rad areas, aligning filters, etc (CFR 43.4)			3.8	98
	Subtotal					2
4. Emergency Procedures / Plan	2.4.6	Knowledge of EOP mitigation strategies. CFR: 43.5			4.7	99
	2.4.40	Knowledge of SRO responsibilities in emergency plan implemetation. CFR: 43.5			4.5	100
	Subtotal					2
Tier 3 Point Total						7

Tier/ Group	Randomly Selected K/A	Reason for Rejection
1 / 1	295021, AA1.03 (3.1)	The current K/A refers to monitoring Component cooling water systems as they apply to a Loss of Shutdown Cooling, CCW system does not interrelate with the RHR Shutdown Cooling system at all at GGNS. Request changing to AA1.05, Reactor Recirculation (3.0)
1 / 1	295038, G2.4.6 (3.7)	The current K/A refers to EOP mitigation strategies associated with High Off-site Release Rate, this K/A is more suited to the SRO level not RO. Request changing to G2.1.27, Knowledge of system purpose and/or function. (3.9)
1 / 2	295020, AK3.07 (3.4)	The current K/A refers to the reason for responses as they apply to Inadvertent Containment Isolation on Suppression Pool temperature response, the Mark III containment at GGNS and the size of the suppression pool will allow a very small change in suppression pool temp on a containment isolation. Request changing to AK 3.01, Reactor SCRAM (3.4)
2 / 1	212000, A1.07 (3.4)	The current K/A is the Ability to predict and/or monitor changes in parameters associated with operating the RPS controls including 'Rod Position information", any change in Rod position information does not affect the RPS system at GGNS. Request changing to A1.01, RPS motor-generator output voltage. (2.8)
2 / 1	215005, K3.02 (3.5)	The current K/A is the knowledge of the effect that a loss of malfunction of the APRM/LPRM system will have on Reactor Recirculation System. The APRM system no longer provides a power feedback to the Reactor Recirc FCV control system. Request changing to K3.01, RPS (3.5)

Tier/ Group	Randomly Selected K/A	Reason for Rejection
2 / 1	264000, G2.1.19 (3.9)	The current K/A is the ability to use computers to evaluate system or component status for the Emergency D/G system, at GGNS the EDGs are not monitored on the PDS plant computer system. Request changing to 2.1.30, Ability to locate and operate components, including local controls. (4.4)
2 / 1	209002, K5.02 (2.6)	The current K/A, inability to develop question without cueing, using the HPCS heat removal mechanism. Request changing to randomly selected K/A K5.04, Adequate core cooling: BWR-5,6
2 / 2	202001, A2.07 (3.1)	The current K/A is the Ability to predict the impacts of "Recirculation pump speed mismatch", GGNS does not have variable speed Recirculation pumps, only Flow Control Valves. Request changing to A2.08, Recirculation Flow mismatch (3.1)
2 / 2	204000, A3.01 (3.2)	The current K/A refers to System pressure downstream of the pressure regulating valve, GGNS does not have a pressure regulating valve but uses pre-pump / post pump modes of operation. Request changing to A3.03, Response to system isolations. (3.6)
3	2.1.7 (4.4)	The current K/A does not easily allow the use of generic knowledge testing. Request changing to 2.1.36, Knowledge of procedures and limitations involved in core alterations. (3.0)

Tier/ Group	Randomly Selected K/A	Reason for Rejection
1 / 2	295013, 2.2.21 (4.1)	The current K/A has a low validity regarding Operation decisions. Request changing to 2.2.40, Ability to apply Technical Specifications for a system. (4.7)
2 / 2	201001, 2.2.38 (4.5)	The current K/A is not included within the facility license at a SRO level. Request changing to 290003, Control Room HVAC, 2.2.38 (4.5)

Facility: **GRAND GULF NUCLEAR STATION**		Date of Examination: **5/8/2017**
Examination Level: RO ☐ SRO ☒		Operating Test Number: **2017U**

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R; M	Determine Fire Watch Requirements GJPM-OPS-2017AS11 2.1.8 (3.4/4.1); 2.1.25(3.9/4.2); 2.2.22 (4.0/4.7); 2.4.25 (3.3/3.7)
Conduct of Operations	R; N	Review work hours and verify Fatigue rules met. GJPM-OPS-2017AS2 2.1.5 (2.9/3.9)
Equipment Control	R; D	Determine Impact on Plant Operations for Failed Relay GJPM-OPS-2017AS33 2.2.41 (3.5/3.9); 2.2.36 (3.1/4.2); 2.2.22 (4.0/4.7)
Radiation Control	R; M	Radioactive Discharge Permit GJPM-OPS-2017AS4 2.3.6 (2.0/3.8)
Emergency Plan	R; D	PAR Determination GJPM-OPS-2017AS5 2.4.44 (2.4/4.4)

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; ≤ 4 for SROs & RO retakes)
(N)ew or (M)odified from bank (≥ 1)
(P)revious 2 exams (≤ 1; randomly selected)

Facility: **GRAND GULF NUCLEAR STATION**		Date of Examination: 05/08/2017
Exam Level: RO ☐ SRO-I ☐ SRO-U ☒		Operating Test No.: LOT-2017U

Control Room Systems@ (8 for RO); (7 for SRO-I); (2 or 3 for SRO-U, including 1 ESF)

System / JPM Title	Type Code*	Safety Function
a. 202001 A4.01 (3.7/3.7), Transfer Recirc Fast to Slow, GJPM-OPS-2017S1	A-M-S	1
b. 223001 A2.11 (3.6/3.8), Manually Initiate Suppression Pool Makeup GJPM-OPS-2017S22	A-D-EN-S	5
c. 205000 A4.01 (3.7/3.7), Startup Shutdown Cooling B, GJPM-OPS-2017S3	D-L-S	4
d. N/A		
e. N/A		
f. N/A		
g. N/A		
h. N/A		

In-Plant Systems@ (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)

System / JPM Title	Type Code*	Safety Function
i. 264000 Generic 2.1.30 (4.4/4.0), Local Start Standby DG 11/12 GJPM-OPS-2017P11	A-D	6
j. 286000 Generic 2.1.30 (4.4/4.0), SBGT Fire Suppression initiation, GJPM-OPS-2017P2	D-E-R	8
k. N/A		

@ All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.

* Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path	4-6 / 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 (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	

Facility: Grand Gulf Nuclear Station Scenario No.: 2 Op-Test No.: NRC LOT 5-2017

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Event No.	Malf. No.	Event Type †	Event Description
1	b21059o lo-1b21f047d_r	C (BOP) A (Crew) TS (CRS)	B21-F047D, Lo-Lo Set SRV failing open (ARI 04-1-02-1H13-P601-18A-G2, 19A-A5, 04-1-02-1H13-P870-3A-E4 & 9A-E4, ONEP 05-1-02-V-21) (TS 3.6.1.6)
2	e22053 e22159a	C (BOP) A (Crew) TS (CRS)	Inadvertent HPCS initiation with failure of E22-F004, HPCS Injection Valve (ARI 04-1-02-1H13-P601-16A-B2, C3 & 5A, 02-S-01-43, section 6.6.2) (TS 3.5.1 Condition B)
3	p43f505	I (ATC)	EHC Temperature Controller failure resulting in N32-F505 closing. (ARI 04-02-1H13-P680-10A-B5, EN-OP-115, section 5.4)
4	tc081	M (CREW)	Pressure Control (IPC) Oscillations – Reactor Scram (ONEP 05-1-02-V-21, EP-2,)
5	rr063a	M (CREW)	Recirc Loop A rupture (EP-2 & EP-3)
6	r21221	I (CREW)	RHR B/C logic power failure (ARIs 04-1-02-1H13-P601-17A-H2, H3 & H6; 04-1-02-1H13-P601-21A-H5)
7	r21139b fw115a	C (CREW)	Bus 14AE trip with Condensate Pump A trip (05-1-02-I-4, Loss of AC Power)

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

Quantitative Attributes Table

Normal Events	0	EOP Contingency Procedures Used	2
Total Malfunctions	7	Simulator Run Time	60
Malfunctions After EOP Entry (1 – 2)	2	EOP Run Time	30
Abnormal Events (2 – 4)	2	Critical Tasks (EOP based 2 – 3)	2
Major Transients (1 – 2)	2	Instrument/Component Failures	5
EOPs Used (Requiring measurable action) (1 – 2)	2	Reactivity Manipulations	0

Objectives: To evaluate the candidates' ability to operate the facility in response to the following evolutions:

1. Respond to B21-F047D, Lo-Lo Set SRV failing open
2. Respond to a spurious HPCS initiation with E22-F004, HPCS Injection Valve, mechanically bound
3. Respond to the EHC Temperature controller failure resulting in N32-F505, EHC Temperature Control Valve, closing.
4. Respond to Main Turbine Pressure Control (IPC) oscillations
5. Respond to Reactor Recirc piping rupture
6. Respond to RHR B/C logic power failure
7. Respond to Bus 14AE trip with Condensate Pump A trip

Initial Conditions: Plant is operating at 90% power.

Inoperable Equipment: None

Turnover:

- Plant is operating at 90% power following a Control Rod sequence exchange.
- Step 12.8 of IOI-2, Attachment VIII, is complete
- Power was held at 90% due to preconditioning restraints.
- Reactor Engineering has determined there are no further power restraints to reach 100% power.
- Immediately following turnover:
  - Raise reactor power to 100% power
- This is a Division 1 work week

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 60 minutes

SCENARIO ACTIVITIES:

B21-F047D, Lo-Lo Set SRV failing open *(This is a new event)*

At the direction of the lead evaluator, **trigger Event 1** to cause B21-F047D, Lo-Lo Set SRV to fail open.

A. The crew will respond using ONEP 05-1-02-V-21, Reactor Pressure Control Malfunctions.

B. The BOP will attempt to close B21-F047D by placing B21-F047D handswitches on P601, P631, P150, and P151 to OFF.

   1. B21-F047D will close when handswitch on P631 is placed to OFF.

C. CRS will determine Tech Spec 3.6.1.6 Condition A applies.

D. Depending on Suppression Pool temperature, the crew may place one or two loops of Suppression Pool Cooling inservice.

Inadvertent HPCS initiation with a failure of E22-F004 to open*(This is a new event)*

A. At the direction of the lead evaluator, **trigger Event 2** to cause a HPCS initiation due to a spurious Reactor Level 2 signal.

B. The BOP will perform the following per 02-S-01-43, Transient Mitigation Strategy:

   1. Verify by at least two independent means that the initiation is spurious
   2. Trip HPCS pump as soon as possible
   3. Close E22-F004

C. The BOP will recognize that E22-F004 has lost power by observing the HPCS MOV OVERLD/PWRLOSS status light and the HPCS OOSVC annunciator illuminated

D. CRS will determine that Tech Spec 3.5.1 Condition B applies.

E. CRS will direct re-alignment of RHR to LPCI Standby lineup.

EHC Temperature Controller failure*(This is a new event)*

A. At the direction of the lead evaluator, **trigger Event 3** to cause the EHC Temperature Controller, N32-R613, to fail resulting in EHC Temperature Valve N32-F505 to close.

B. The crew will respond in accordance with EN-OP-115, section 5.4 and ARI 04-1-02-1H13-P680-10A-B5.

C. The ATC will take manual control of controller N32-R613 and restore EHC temperature.

Pressure Control Oscillations*(This is a new event based on CR-GGN-2016-4834, Operator's Decision Prior to an Automatic Scram)*

A. At the direction of the lead evaluator, **trigger Event 4** to cause Initial Pressure Control (IPC) oscillations.

B. The crew will respond in accordance with ONEP 05-1-02-V-21, Reactor Pressure Control Malfunctions.

C. The crew will observe APRM oscillations and upon determining that oscillations will exceed or have exceeded 10 %, place the Reactor Mode Switch to SHUTDOWN in accordance with ONEP 05-1-02-V-21.

D. The crew will enter Scram and Turbine Trip ONEP and EP-2.

Recirc Loop A suction line break in Drywell *(This is different event from 2014 NRC scenario 3 due to the available injection sources and different operator actions required.)*

A. When the reactor is scrammed, **Auto trigger Event 5** will cause a Recirc Loop A suction rupture after 5 minutes.

B. Reactor water level will lower and Drywell temperature and pressure will rise due to the leak.

C. The crew will enter EP-3 on Drywell temperature and pressure.

RHR B/C Logic Power failure *(This is a new event.)*

A. When Drywell pressure reaches 1.39 psig, RHR B and C logic power will be lost (**Auto Event 6**).

B. No Division 2 LOCA signal will be received by LŠS, Div 2 Diesel Generator, ADS B, CGCS B or SSW B.

C. RHR B pump can be manually started, but the injection valve must be manually opened from the Remote Shutdown Panel (per ARI P601-17A-H2, RHR B SYS OOSVC, step 4.6).

   1. If directed, open E12-F042B, RHR B Injection Valve, from the Remote Shutdown Panel using the panel mimic for 1H22-P151.

D. RHR C pump can be manually started, but the injection valve must be manually opened locally.

   1. If directed, **trigger Event 8** to manually open E12-F042C, RHR C Injection Valve.

E. E12-F064B and E12-F064C, RHR B and C Minimum Flow valves, will not open or close automatically.

Bus 14AE trip with Condensate Pump A trip *(This is a new event)*

A. Ten minutes after the reactor scram, **auto Event 7** will cause BOP Bus 14AE to trip/lockout and Condensate Pump A will trip, resulting in a total loss of feedwater to the Reactor.

B. Reactor water level will lower due to the Recirc Loop leak being greater than the capability of RCIC and CRD.

C. Reactor water level will lower to < -160" requiring an Emergency Depressurization.

D. Crew will restore reactor level with available ECCS systems.

Termination:

A. Once emergency depressurization has been conducted and reactor water level is stabilized above TAF, or as directed by Lead Evaluator:

   1. Take the simulator to Freeze and turn horns off.
   2. Stop and save the SBT report and any other recording devices.
   3. Instruct the crew to not erase any markings or talk about the scenario until after follow-up questions are asked.

Critical Task		
Number	Description	Basis
1	* **Open 7 SRVs prior to RPV level reaching -191 inches compensated fuel zone.**	If an injection source is available but the decreasing RPV water level trend cannot be reversed before RPV water level drops to the Minimum Steam Cooling RPV Water Level (-191 in.), emergency RPV depressurization is required to permit injection from low head systems, maximize flow from available injection sources, and minimize the flow through any primary system break. (per 02-S-01-40, EP Technical Bases)
2	* **After Emergency Depressurization, restore and maintain RPV level above -191" using available injection systems prior to exiting EP-2.**	The Minimum Steam Cooling RPV Water Level is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. (per 02-S-01-40, EP Technical Bases)
*	**Critical Task (As defined in NUREG 1021 Appendix D), If an operator or the crew significantly deviates from or fails to follow procedures that effect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review.**	

Facility: Grand Gulf Nuclear Station Scenario No.: 3 Op-Test No.: NRC LOT 5-2017

Examiners: ____________ Operators: ____________

Event No.	Malf. No.	Event Type †	Event Description
1	di_1b33k603bc	C (ATC) TS CRS) A (CREW)	Recirc FCV B fails closed. (EN-OP-115, Reduction in Recirc ONEP, Tech Spec 3.4.1)
2	r21139f n41140b	C (BOP) A (CREW) TS (CRS)	Bus 16AB trip with failure of Div 2 Diesel Generator to start (Tech Spec 3.8.1 Condition B and 05-1-02-I-4, Loss of AC Power ONEP)
3	r21142aa	TS(CRS) A (CREW)	LCC 16BB4 fails to re-energize (Tech Spec 3.8.7 Condition A, 3.7.1, Condition and ONEP 05-1-02-I-4, Loss of AC Power)
4	rr011e	C (ATC) TS (CRS) A (CREW)	Jet Pump 9/10 failure (Jet Pump Anomalies ONEP, 05-1-02-III-6, Tech Spec 3.4.3, 3.4.1)
5	ct218d ct219a	M (CREW)	Suppression Pool leak in LPCS suction piping with failure of LPCS Pump Room watertight door – Failure of Suppression Pool Makeup – Reactor Scram. (EP-3, EP-4, EP-2)
6	r21133a	C (CREW)	Loss of Service Transformer 11 with failure of Div 1 Diesel Generator to start (Loss of AC Power ONEP, Loss of Feedwater ONEP, EP-2, EP-3)
7	e22052	C (CREW)	HPCS Pump trip (EP-2)
8	r21134h	C (CREW)	Loss of ESF 12 Transformer (Loss of AC Power ONEP)

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

Quantitative Attributes Table

Normal Events	0	EOP Contingency Procedures Used	2
Total Malfunctions	8	Simulator Run Time	60
Malfunctions After EOP Entry (1 – 2)	3	EOP Run Time	30
Abnormal Events (2 – 4)	4	Critical Tasks (EOP based 2 – 3)	2
Major Transients (1 – 2)	1	Instrument/Component Failures	6
EOPs Used (Requiring measurable action) (1 – 2)	3	Reactivity Manipulations	0

Objectives: To evaluate the candidates' ability to operate the facility in response to the following evolutions:

1. Respond to Recirc FCV B failing closed.
2. Respond to a trip of ESF Bus 16AB Feeder Breaker 152-1614 with a failure of Division 2 Diesel Generator to start.
3. Respond to a failure of LCC 16BB4 to re-energize.
4. Respond to a Jet Pump 9/10 failure.
5. Respond to a Suppression Pool leak in the LPCS suction piping with a failure of the LPCS Pump Room watertight door.
6. Respond to a loss of Service Transformer 11 with a failure of Division 1 Diesel Generator to start.
7. Respond to a loss of ESF Transformer 12.

Initial Conditions: Plant is operating at 100% power.

Inoperable Equipment: NONE

Turnover:

- Plant is operating at 100% power.
- This is a Division 1 work week
- RCIC Trip/Throttle is tagged OOS for work on the overspeed trip linkage.
- Potential thunderstorms are forecasted for the site during this shift.

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 60 minutes

SCENARIO ACTIVITIES:

Recirc FCV B failing closed. *(This is a new event)*

A. At the direction of the lead evaluator, **trigger Event 1** to cause Recirc FCV B to slowly close.

B. The crew will recognize the Recirc FCV B closing and trip the Hydraulic Power Unit to stop Recirc FCV B motion.

C. The crew will verify Recirc flow mismatch with Technical Specification limit of 2230 gpm. If mismatch is greater than 2230 gpm, the crew will lower flow in Recirc Loop A using SOI 04-1-01-B33-1 until flows are within the limit.

D. If Recirc flows are not within the Tech Spec limit of 2230 gpm, the CRS will recognize that Tech Spec 3.4.1, Condition A is applicable.

Bus 16AB Feeder Breaker trip with failure of Division 2 Diesel Generator to start *(This event is different from previous malfunctions due to the operator's ability to restore Bus 16AB from a different offsite source (ESF 12)*

A. At the direction of the lead evaluator, **trigger Event 2** to cause the breaker 152-1614, BUS 16AB FDR FM ESF XFMR 21 to trip.

B. The crew will respond in accordance with ONEP 05-1-02-I-4, Loss of AC Power.

C. The BOP will recognize the Division 2 Diesel Generator did not automatically start and will manually re-energize Bus 16AB from ESF Transformer 12 by closing breaker 152-1611.

D. CRS will recognize Tech Spec 3.8.1, Condition B is applicable.

LCC 16BB4 fails to re-energize *(This is a repeat event from NRC 2014 Scenario 3. This event is added to assist in the failure of Suppression Pool Makeup System in Event 6. This event contains no discernable operator actions other than Tech Spec entry)*

A. The BOP will recognize the failure of LCC 16BB4 to re-energize.

B. The crew will respond in accordance with ONEP 05-1-02-I-4, Loss of AC Power.

C. CRS will recognize Tech Spec 3.8.7, Condition A is applicable.

Jet Pump 9/10 Failure *(This is a new event)*

A. At the direction of the lead evaluator, **trigger Event 4** to cause a failure of the rams head for Jet Pumps 9 & 10.

B. The crew will respond in accordance with Jet Pump Anomalies ONEP, 05-1-02-III-6.

C. CRS will contact Reactor Engineering to perform 06-RE-1B33-D-0001, Jet Pump Functional Test.

D. CRS will direct the BOP to calculate total core flow in accordance with NOTE for step 3.5 of Jet Pump Anomalies ONEP.

E. CRS will recognize Tech Spec 3.4.3, Condition A, is applicable for the failed jetpumps

F. CRS will recognize Tech Spec 3.4.1, Condition A is applicable for Recirc loop flow mismatch.

Suppression Pool leak in LPCS suction piping with failure of LPCS Pump Room watertight door *(This is a new event)*

A. At the direction of the lead evaluator, **trigger Event 5** will cause LPCS suction piping rupture.

B. Suppression Pool level will lower and LPCS Room sump level HI-HI annunciator will alarm.

C. The crew will enter EP-4 on Area Water Level above the Operating Limit and EP-3 on Suppression Pool Level below 18.34 ft.

D. The crew will attempt to isolate the leak by closing E21-F001, LPCS PMP SUCT FM SUPP POOL.

E. The crew will attempt to initiate Suppression Pool Makeup System, per EP-3. Suppression Pool Makeup System will not initate.

F. Suppression Pool level will continue to lower.

G. Crew will enter EP-2 and manually scram the reactor.

H. Suppression Pool level will continue to lower, requiring Emergency Depression of the reactor when it is determined that Suppression Pool level cannot be maintained above 14.5 feet.

I. Crew will emergency depressurize the reactor using SRVs before Suppression Pool Level lowers to 14.5 feet.

J. Crew will restore reactor level with available systems.

Loss of Service Transformer 11 *(This is a new event)*

A. When the Reactor Mode Switch is placed in SHUTDOWN, **auto trigger Event 6** will cause a loss of Service Transformer 21.

B. Crew will recognize the Division 1 Diesel Generator did not automatically start and will manually re-energize Bus 15AA from ESF Transformer 21 by closing breaker 152-1501.

C. Crew will restore power to BOP Bus 12AE and 13AD and restore Condensate/Feedwater.

B. Crew may initiate HPCS and maximize CRD to maintain reactor level.

C. Crew will identify HPCS trip, if initiated.

Loss of ESF Transformer 12 *(This is a new event)*

A. 5 minutes after the Reactor Mode Switch is placed in SHUTDOWN, ESF Transformer 12 will lockout.

B. The crew will recognize the Div 2 Diesel Generator is inoperable and attempt to restore power from ESF Transformer 21, but the breaker 152-1614 will not close.

Termination:

A. Once reactor depressurization has been conducted and reactor water level is stabilized above TAF, or as directed by Lead Evaluator:

1. Take the simulator to Freeze and turn horns off.
2. Stop and save the SBT report and any other recording devices.
3. Instruct the crew to not erase any markings or talk about the scenario until after follow-up questions are asked.

Critical Task		
Number	Description	Basis
1	* **Open 7 SRVs prior to Suppression Pool level reaches 14.5 feet.**	Suppression Pool level must be maintained above 14.5 feet to ensure that steam discharged through the horizontal vents following a primary system break will be adequately condensed. If a primary system break were to occur with suppression pool water level below this elevation, pressure suppression capability would be unavailable and primary containment pressure could exceed structural limits. If suppression pool water level cannot be maintained above 14.5 feet, emergency RPV depression is required since the RPV is not permitted to remain at pressure if pressure suppression capability unavailable. (per 02-S-01-40, EP Technical Bases)
2	* **Restore injection source to the RPV prior to Emergency Depressurization.**	Maintain adequate core cooling.

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

Facility: Grand Gulf Nuclear Station Scenario No.: 1 Op-Test No.: NRC LOT 5-2017

Examiners: ______________ Operators: ______________

Event No.	Malf. No.	Event Type †	Event Description
1	rci036	C (ATC)	Control Rod 28-53 GN data fault resulting in a control rod withdrawal block (Alarm Response Instruction 04-1-02-1H13-P680-4A2-C5; 04-1-01-C11-2, Rod Control and Information System, section 4.11).
2	tte31n043b_a g33f251	TS (CRS) C (BOP) A (CREW)	Division 2 RWCU Isolation due to failed temperature switch with G33-F251, RWCU SPLY TO RWCU HXS, power loss. (05-1-02-III-5, Automatic Isolation ONEP, TS 3.6.5.3 Condition A, TS 3.3.6.1, Condition A)
3	pte22n654c_a	TS (CRS) C (BOP)	E22-N654C, CST LEVEL LO trip unit fails high (Tech Spec 3.3.5.1, Condition D)
4	fw124	C(ATC) A (CREW)	Failure of Startup Level Control Valve (05-1-02-V-8, Feedwater Malfunctions ONEP)
5	p680_2a_e_9 fw115a, b, & c	M (CREW)	Respond to a false Hotwell low level signal – Total Loss of Feedwater – Reactor Scram (Alarm Response Instruction 04-1-03-P680-2A-E9, EP-2)
6	c11167	M (CREW)	Low power ATWS – Failure to Vent (EP-2A)
7	c41263	C (CREW)	Standby Liquid Control Piping Rupture
8	e51043 di_1e51M625D e51044	C (CREW)	Failure of RCIC to manually initiate (manual start available) (SOI 04-1-01-E51-1, Reactor Core Isolation Cooling System)

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

Quantitative Attributes Table

Normal Events	0	EOP Contingency Procedures Used	1
Total Malfunctions	8	Simulator Run Time	90
Malfunctions After EOP Entry (1 – 2)	2	EOP Run Time	45
Abnormal Events (2 – 4)	2	Critical Tasks (EOP based 2 – 3)	2
Major Transients (1 – 2)	2	Instrument/Component Failures	5
EOPs Used (Requiring measurable action) (1 – 2)	1	Reactivity Manipulations	0

Objectives: To evaluate the candidates' ability to operate the facility in response to the following evolutions:

1. Withdraw Control Rods to 10% Bypass Valve position
2. Respond to a Control Rod Data Fault on Control Rod 28-53.
3. Respond to a Division 2 RWCU Isolation with G33-F251power loss.
4. Respond to a gross fail high of trip unit E22-N654C, CST Level LO
5. Respond to Startup Level Control Valve actuator rupture
6. Respond to failure of Hotwell level transmitter
7. Take action for Failure to Vent ATWS (< 5% power)
8. Respond to RCIC failure to manually initiate (manual start available)

Initial Conditions:

- Reactor startup in progress
- Reactor power is approximately 4%
- Reactor pressure is 750 psig.

Inoperable Equipment: None

Turnover:

- A reactor startup is in progress
  - Step 90 of Control Rod Movement Sequence is complete
  - SJAE B is in service
  - Step 44 of Attachment XV in 03-1-01-1
- Condensate System is lined up as follows:
  - Condensate Pumps B and C in service
  - Condensate Booster Pump A in service
  - Reactor Feed Pump A in service at approximately 970 psig discharge pressure
  - CFFF is in service
  - 4 Deep Bed Demins are in service
- Annunciators P680-4A2-C5, CONT ROD WITHDRAWL BLOCK, and P680-4A1-A7, CRD DRIVE WTR TO RX ΔP HI, are flagged as expected annunciators.
- Withdraw controls until 10% Bypass Control Valve position, then continue raising TURB STM PRESS DEMAND setpoint to 935 psig.

Scenario Notes:

This is a Low Power Scenario

This scenario is a NEW Scenario.

Validation Time: 60 minutes

SCENARIO ACTIVITIES:

Withdraw control rods per control rod movement sequence to establish Main Turbine Bypass Control Valves 10% open *(This is a repeat evolution from 2014 NRC Scenario 5, but is necessary to allow Event 1)*

A. The crew will withdraw control rods per the startup rod sequence. When control rod 28-53 GN, the second control rod of the pull sheet to be performed, is withdrawn past position 04, a control rod block due to a simulated failed rod position reed switch will occur **(Auto Event 1).**

Control Rod 28-53 Data Fault *(This is a new event based on CR-GGN-2016-3001, Mispositioned Control Rod)*

A. The crew will respond using ARI 04-1-02-1H13-P680-4A2-C5, CONT ROD WITHDRAWL BLOCK and 04-1-01-C11-2, Rod Control and Information System, section 4.11.

B. If contacted as I&C to investigate the data fault, respond there is nothing that can be determined from the Control Room and that a WO will be needed for any troubleshooting from cabinets inside Containment.

Division 2 RWCU Isolation due to failed temperature switch with G33-F251, RWCU SPLY TO RWCU HXS, power loss *(This is a new event)*

A. At the direction of the lead evaluator, **trigger Event 2** to cause temperature switch E31-N623B, RWCU VALVE NEST ROOM, to fail upscale, causing a Division 1 RWCU isolation.

B. The crew will respond using ARI 04-1-02-1H13-P680-11A-A3, RWCU HX MR TEMP HI/INOP and 05-1-02-III-5, Automatic Isolation ONEP.

C. Crew will recognize G33-F251, RWCU SPLY TO RWCU HXS, power loss.

D. CRS will recognize Tech Spec 3.6.5.3, Condition A and 3.3.6.1 Condition A applies.

E. CRS will direct closure of G33-F250, RWCU SPLY TO RWCU HXS, per Tech Spec 3.6.5.3.

E22-N654C, CST LEVEL LO trip unit fails high *(This is a new event)*

A. At the direction of the lead evaluator, **trigger Event 3** to cause E22-N654C, CST LEVEL LO, to gross fail high.

B. The BOP will recognize and report a HPCS SYS OOSVC alarm and TRIP UNIT IN CAL/GR FAIL status light.

C. The BOP will respond to the back panel area and determine that trip unit E22-N654C indicates gross fail high.

D. If contacted as I&C to investigate gross fail, respond that a work request will be generated to start troubleshooting.

E. CRS will determine Tech Spec 3.3.5.1 Condition A applies and, using Table 3.3.5.1-1, enter Condition D.

F. CRS will direct the BOP to swap HPCS suction to the Suppression Pool.

G. BOP will open E22-F015, HPCS Suction from Suppression Pool and close E22-F001, HPCS Suction from CST.

SULC Valve Actuator Failure with N21-F040 power loss*(This is different from event in 2014 NRC scenario 5 due to failure being the valve actuator, not the level controller, requiring the use of the Startup Level Control Bypass Valve, N21-F040.)*

A. At the direction of the lead evaluator, **trigger Event 4** to cause a failure of the Startup Level Control Valve actuator, resulting in the valve rapidly failing closed.

B. The crew will respond to a Startup Level Control Valve failing closed by entering the Feedwater System Malfunctions ONEP.

C. The ATC will take manual control of the Startup Level Control valve controller to maintain reactor water level.

D. The ATC will open the N21-F040, Startup Level Control Bypass Valve, to control reactor water level.

E. If a manual or automatic scram occurs, the crew will enter EP-2 and the Reactor Scram ONEP.

False Hotwell Level Low*(This is a duplicate event from 2015 NRC Scenario 2, however, the crew will be able to restore feedwater to the reactor vessel using the Alarm Response Instruction before an emergency depressurization is required.)*

A. At the direction of the lead evaluator, **trigger Event 5** to cause a false Hotwell level low signal, resulting in the trip of all Condensate pumps, which will cause a trip of all Condensate Booster Pumps and the running Feedwater pump, resulting a loss of feedwater to the reactor vessel.

B. If a scram has not occurred, the loss of feedwater will result in a reactor scram.

C. Crew will enter EP-2 and the Reactor Scram ONEP and initiate RCIC per Feedwater Malfunction ONEP.

D. If requested, **trigger Event 10** can be used to simulate removal of relay N19 63X-1/N105, allowing the Condensate Pumps, Condensate Booster Pumps, and Reactor Feedwater Pumps to be restarted.

   1. Do not insert **trigger 10** until the crew has begun driving in control rods.

Failure of Scram Air Header to Vent ATWS (<5% power) *(This is different than other previous used ATWS malfunctions due to type of ATWS, a failure of the Scram Air Header to vent.)*

The crew will recognize control rods fail to fully insert due to failure of the scram air header to vent. **(Event 6)**.

A. The CRS will enter EP-2A.

B. Insert EP Attachments as requested:

The following can be used to insert the EP Attachments

1. Attachment 12 – trigger Event 12
2. Attachment 18 – trigger Event 13
3. Attachment 19 – trigger Event 14
4. Attachment 20 – trigger Event 15
5. Attachment 23 – trigger Event 16 **(do not insert until at least 4 gangs of control rods are inserted or at the direction of the lead evaluator)**
6. Attachment 8 – trigger Event 17

RCIC Failure to Initiate (manual start available) *(This is a duplicate event from 2014 NRC Scenario 5, however, RCIC will not function following manual start due to speed controller failure. Event is necessary to prevent reactor level restoration prior to inserting control rods.)*

A. The BOP will recognize the failure of the RCIC Manual Initiate pushbuttons to initate RCIC **(Event 7)** and perform a manual start of RCIC per SOI 04-1-01-E51-1 (hard card).

B. The BOP will recognize that RCIC fails to come up to speed and will not be able to inject into the reactor vessel.

Termination:

A. Once all control rods are fully inserted and reactor water level is being controlled in band or as directed by Lead Evaluator:

- Take the simulator to Freeze and turn horns off.
- Stop and save the SBT report and any other recording devices.
- Instruct the crew to not erase any markings or talk about the scenario until after follow-up questions are asked.

Critical Task		
Number	Description	Basis
1	* **Following an ATWS, insert control rods by venting the scram air header and/or normal rod insertion prior to exiting EP-2A.**	Positive confirmation that the reactor will remain shutdown under all conditions is best obtained by verifying that all control rods are inserted to or beyond position 02. Position 02 is the "Maximum Subcritical Banked Withdrawal Position," defined to be the greatest banked rod position at which the reactor will remain shutdown under all conditions. (per 02-S-01-40, EP Technical Bases)
2	* **Restore Condensate/Feedwater and/or RHR A or B injection prior to level reaching TAF (-167" Fuel Zone).**	Maintaining adequate core cooling.
*	**Critical Task (As defined in NUREG 1021 Appendix D), If an operator or the crew significantly deviates from or fails to follow procedures that effect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review.**	