Form 4.1-BWR Pressurized-Water Reactor Examination Outline

Facility: Grand Gulf 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	4	4				3	3			3	20			
	2	1	1	1	N/A			1	1	N/A		1	6			
	Tier Totals	4	5	5				4	4			4	26			
2. Plant Systems	1	2	2	3	3	3	3	2	2	2	2	2	26			
	2	1	1	1	1	1	1	1	1	1	1	1	11			
	Tier Totals	3	3	4	4	4	4	3	3	3	3	3	37			

3. Generic Knowledge and Abilities Categories	CO	EC	RC	EM	Total	SRO CO	SRO EC	SRO RC	SRO EM	Total
	2	2	1	1	6					

4. Theory	Reactor Theory	Thermodynamics	Total
	3	3	6

Notes: CO = Conduct of Operations; EC = Equipment Control; RC = Radiation Control; EM = Emergency Procedures/Plan

* These systems/evolutions may be eliminated from the sample when Revision 2 of the K/A catalog is used to develop the sample plan

** These systems/evolutions are only included as part of the sample (as applicable to the facility) when Revision 2 of the K/A catalog is used to develop the sample plan

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			X				Knowledge of the reasons for the following responses or actions as they apply to Partial or Complete Loss of Forced Core Flow Circulation: (CFR: 41.5 / 45.6) AK3.07 Recirculation pump discharge/suction valve manipulation	3.3	14
295003 (APE 3) Partial or Complete Loss of AC Power	X						Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Partial or Complete Loss of AC Power: (CFR: 41.8 to 41.10) AK1.01 Battery capacity	3.8	8
295004 (APE 4) Partial or Total Loss of DC Power				X			Ability to operate and/or monitor the following as they apply to Partial or Complete Loss of DC Power: (CFR: 41.7 / 45.6) AA1.03 AC electrical distribution	3.5	12
295005 (APE 5) Main Turbine Generator Trip			X				Knowledge of the reasons for the following responses or actions as they apply to Main Turbine Generator Trip: (CFR: 41.5 / 45.6) AK3.04 Main generator trip	3.2	15
295006 (APE 6) Scram						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	20
295016 (APE 16) Control Room Abandonment / 7						X	2.4.34 Knowledge of RO responsibilities outside the main control room during an emergency (CFR: 41.10 / 43.5 / 45.13)	4.2	10
295018 (APE 18) Partial or Complete Loss of CCW		X					Knowledge of the relationship between Partial or Complete Loss of Component Cooling Water and the following systems or components: (CFR: 41.7 / 45.8) AK2.04 Reactor recirculation system	3.7	13
295019 (APE 19) Partial or Complete Loss of Instrument Air		X					Knowledge of the relationship between Partial or Complete Loss of Instrument Air and the following systems or components: (CFR: 41.7 / 45.8) AK2.23 Service water	2.7	2
295021 (APE 21) Loss of Shutdown Cooling					X		Ability to determine and/or interpret the following as they apply to Loss of Shutdown Cooling: (CFR: 41.10 / 43.5 / 45.13) AA2.02 RHR/shutdown cooling system flow	4.1	19
295023 (APE 23) Refueling Accidents			X				Knowledge of the reasons for the following responses or actions as they apply to Refueling Accidents: (CFR: 41.5 / 45.6) AK3.03 Ventilation isolation	3.7	1
295024 (EPE 1) High Drywell Pressure	X						Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to High Drywell Pressure: (CFR: 41.8 to 41.10) EK1.02 Containment building integrity (Mark III)	4.0	9
295025 (EPE 2) High Reactor Pressure				X			Ability to operate and/or monitor the following as they apply to High Reactor Pressure: (CFR: 41.7 / 45.6) EA1.01 Main and reheat steam	3.1	11

295026 (EPE 3) Suppression Pool High Water Temperature			X				Knowledge of the reasons for the following responses or actions as they apply to Suppression Pool High Water Temperature: (CFR: 41.5 / 45.6) EK3.04 SLCS injection	3.6	17
295027 (EPE 4) High Containment Temperature (Mark III Containment Only)						X	2.4.3 Ability to identify post-accident instrumentation (CFR: 41.6 / 45.4)	3.7	7
~~295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only)~~									
295030 (EPE 7) Low Suppression Pool Water Level				X			Ability to operate and/or monitor the following as they apply to Low Suppression Pool Water Level: (CFR: 41.7 / 45.6) EA1.01 ECCSs	4.0	5
295031 (EPE 8) Reactor Low Water Level	X						Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Reactor Low Water Level: (CFR: 41.8 to 41.10) EK1.01 Adequate core cooling	4.7	6
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown		X					Knowledge of the relationship between SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown and the following systems or components: (CFR: 41.7 41.8 / 45.8) EK2.12 RCIS (BWR 6)	3.7	18
295038 (EPE 15) High Offsite Radioactivity Release Rate					X		Ability to determine and/or interpret the following as they apply to High Offsite Radioactivity Release Rate: (CFR: 41.10 / 43.5 / 45.13) EA2.02 Total number of curies released or release rate/duration	3.2	3
600000 (APE 24) Plant Fire on Site		X					Knowledge of the relationship between Plant Fire on Site and the following systems or components: (CFR 41.7 / 45.7) AK2.07 Electrical distribution system	3.3	4
700000 (APE 25) Generator Voltage and Electric Grid Disturbances					X		Ability to determine and/or interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: (CFR: 41.5 and 43.5 / 45.5 / 45.7 / 45.8) AA2.06 Generator frequency limitations	3.2	16
K/A Category Totals:	3	4	4	3	3	3	Group Point Total:		20

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									
295007 (APE 7) High Reactor Pressure									
295008 (APE 8) High Reactor Water Level		X					Knowledge of the relationship between High Reactor Water Level and the following systems or components: (CFR: 41.7 / 45.8) AK2.04 PCIS/NSSSS	3.6	23
295009 (APE 9) Low Reactor Water Level						X	2.1.45 Ability to identify and interpret diverse indications to validate the response of another indication (CFR: 41.7 / 43.5 / 45.4)	4.3	25
295010 (APE 10) High Drywell Pressure	X						Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to High Drywell Pressure: (CFR: 41.8 to 41.10) AK1.03 Drywell temperature increase	3.8	22
295011 (APE 11) High Containment Temperature (Mark III Containment only)					X		Ability to determine and/or interpret the following as they apply to High Containment Temperature: (CFR: 41.10 / 43.5 / 45.13) AA2.02 Containment pressure	3.8	24
295012 (APE 12) High Drywell Temperature									
295013 (APE 13) High Suppression Pool Water Temperature/ 5									
295014 (APE 14) Inadvertent Reactivity Addition				X			Ability to operate and/or monitor the following as they apply to Inadvertent Reactivity Addition: (CFR: 41.7 / 45.6) AA1.10 HPCS	3.5	26
295017 (APE 17) High Offsite Release Rate									
295020 (APE 20) Inadvertent Containment Isolation									
295022 (APE 22) Loss of Control Rod Drive Pumps									
295029 (EPE 6) High Suppression Pool Water Level									
295032 (EPE 9) High Secondary Containment Area Temperature									
295033 (EPE 10) High Secondary Containment Area Radiation Levels									
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9									
295035 (EPE 12) Secondary Containment High Differential Pressure									
295036 (EPE 13) Secondary Containment High Sump/Area Water Level									
500000 (EPE 16) High Containment Hydrogen Concentration			X				Knowledge of the reasons for the following responses or actions as they apply to High Containment Hydrogen Concentration: (CFR: 41.5 / 45.6) EK3.10 Operation of the hydrogen igniters (BWR 6)	3.6	21
K/A Category Point Totals:	1	1	1	1	1	1	Group Point Total:		6

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 RHR/LPCI: Injection Mode design features and/or interlocks that provide for the following: (CFR: 41.7) K4.06 No-suction path pump trip	3.5	51
205000 (SF4 SCS) Shutdown Cooling	X											Knowledge of the physical connections and/or cause and effect relationships between the Shutdown Cooling System (RHR Shutdown Cooling Mode) and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.18 Feedwater system (BWR 6)	3.5	35
~~206000 (SF2, SF4 HPCI) High-Pressure Coolant Injection~~														
~~207000 (SF4 IC) Isolation (Emergency) Condenser~~														
209001 (SF2, SF4 LPCS) Low-Pressure Core Spray							X					Ability to predict and/or monitor changes in parameters associated with operation of the Low-Pressure Core Spray System, including: (CFR: 41.5 / 45.5) A1.07 Emergency generator loading	3.9	50
209002 (SF2, SF4 HPCS) High-Pressure Core Spray										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.14 Test return valve	3.3	34
211000 (SF1 SLCS) Standby Liquid Control									X			Ability to monitor automatic operation of the Standby Liquid Control System, including: (CFR: 41.7 / 45.7) A3.08 System initiation	4.1	44
212000 (SF7 RPS) Reactor Protection	X											Knowledge of the physical connections and/or cause and effect relationships between the Reactor Protection System and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.06 Control rod drive hydraulic system	3.9	40
215003 (SF7 IRM) Intermediate-Range Monitor		X										Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 IRM channels/detectors	3.4	29
215004 (SF7 SRMS) Source-Range Monitor			X									Knowledge of the effect that a loss or malfunction of the Source Range Monitor System will have on the following systems or system parameters: (CFR: 41.7 / 45.4) K3.03 RCIS (BWR 6)	3.9	28
215005 (SF7 PRMS) Average Power Range Monitor/Local Power Range Monitor								X				Ability to (a) predict the impacts of the following on the Average Power Range Monitor/Local Power Range Monitor System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.6) A2.04 SCRAM trip signals	4.4	36
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling											X	2.1.23 Ability to perform general or normal operating procedures during any plant condition (CFR: 41.10 / 43.5 / 45.2 / 45.6)	4.3	38

218000 (SF3 ADS) Automatic Depressurization										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.05 ADS timer reset	4.0	52
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff									X			Ability to monitor automatic operation of the Primary Containment Isolation System/Nuclear Steam Supply Shutoff, including: (CFR: 41.7 / 45.7) A3.01 System indicating lights and alarms	4.0	39
239002 (SF3 SRV) Safety Relief Valves		X										Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 SRV solenoids	3.7	46
259002 (SF2 RWLCS) Reactor Water Level Control							X					Ability to predict and/or monitor changes in parameters associated with operation of the Reactor Water Level Control System, including: (CFR: 41.5 / 45.5) A1.01 Reactor water level	4.3	49
261000 (SF9 SGTS) Standby Gas Treatment						X						Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Standby Gas Treatment System: (CFR: 41.7 / 45.7) K6.04 Radiation monitoring system	3.4	45
262001 (SF6 AC) AC Electrical Distribution				X								Knowledge of AC Electrical Distribution design features and/or interlocks that provide for the following: (CFR: 41.7) K4.01 Lockouts	3.5	42
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)											X	2.1.27 Knowledge of system purpose and/or function (CFR: 41.7)	3.9	47
263000 (SF6 DC) DC Electrical Distribution					X							Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the DC Electrical Distribution: (CFR: 41.5 / 45.3) K5.01 Hydrogen generation	3.1	32
264000 (SF6 EGE) Emergency Generators (Diesel/Jet)			X									Knowledge of the effect that a loss or malfunction of the Emergency Generators will have on the following systems or system parameters: (CFR: 41.7 / 45.4) K3.04 Bus frequency/voltage	3.9	37
300000 (SF8 IA) Instrument Air					X							Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Instrument Air System: (CFR: 41.5 / 45.3) K5.14 Air leaks	3.3	30
400000 (SF8 CCW) Component Cooling Water						X						Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Component Cooling Water System: (CFR: 41.7 / 45.7) K6.13 Fuel pool cooling and cleanup system	3.3	33
510000 (SF4 SWS) Service Water								X				Ability to (a) predict the impacts of the following on the Service Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 41.1 / 43.5 / 45.3 / 45.6 / 45.13) A2.08 Abnormal intake water level	3.7	48

262001 (SF6 AC) AC Electrical Distribution					X							Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the AC Electrical Distribution: (CFR: 41.5 / 45.3) K5.01 Paralleling AC sources	3.9	43
263000 (SF6 DC) DC Electrical Distribution				X								Knowledge of DC Electrical Distribution design features and/or interlocks that provide for the following: (CFR: 41.7) K4.03 Ground detection	2.9	31
264000 (SF6 EGE) Emergency Generators (Diesel/Jet)						X						Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Emergency Generators: (CFR: 41.7 / 45.7) K6.10 Jacket water components	3.3	41
400000 (SF8 CCW) Component Cooling Water			X									Knowledge of the effect that a loss or malfunction of the Component Cooling Water System will have on the following systems or system parameters: (CFR: 41.7 / 45.6) K3.19 Reactor/turbine pressure regulating system	2.8	27
K/A Category Point Totals:	2	2	3	3	3	3	2	2	2	2	2	Group Point Total:		26

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														
~~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				X								Knowledge of Traversing In-Core Probe design features and/or interlocks that provide for the following: (CFR: 41.7) K4.03 Radiation shielding	2.9	58
~~215002 (SF7 RBMS) Rod Block Monitor~~														
216000 (SF7 NBI) Nuclear Boiler Instrumentation							X					Ability to predict and/or monitor changes in parameters associated with operation of the Nuclear Boiler Instrumentation, including: (CFR: 41.5 / 45.5) A1.07 RPV level	3.8	62
219000 (SF5 RHR SPC) RHR/LPCI: Torus/Suppression Pool Cooling Mode			X									Knowledge of the effect that a loss or malfunction of the RHR/LPCI: Torus/Suppression Pool Cooling Mode will have on the following systems or system parameters: (CFR: 41.7 / 45.4) K3.03 Primary containment	3.9	53
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 Cooling Mode~~														
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup		X										Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 Fuel pool cooling pumps	3.1	59
234000 (SF8 FH) Fuel Handling Equipment														
239001 (SF3, SF4 MRSS) Main and Reheat Steam														
239003 (SF9 MSIVLC) Main Steam Isolation Valve Leakage Control														

241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating	X											Knowledge of the physical connections and/or cause and effect relationships between the Reactor/Turbine Pressure Regulating System and the following systems: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.39 Main and reheat steam system	3.3	57
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary														
256000 (SF2 CDS) Condensate						X						Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Condensate System: (CFR: 41.7 / 45.7) K6.09 Offgas system	2.6	61
259001 (SF2 FWS) Feedwater														
268000 (SF9 RW) Radwaste										X		Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.02 Primary containment sump pumps	3.1	54
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									X			Ability to monitor automatic operation of the Secondary Containment, including: (CFR: 41.7 / 45.7) A3.01 Secondary containment isolation	4.1	55
290003 (SF9 CRV) Control Room Ventilation					X							Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Control Room Ventilation: (CFR: 41.5 / 45.3) K5.01 Control of airborne contamination (e.g., radiological, toxic gas, smoke)	3.6	63
290002 (SF4 RVI) Reactor Vessel Internals											X	2.1.28 Knowledge of the purpose and function of major system components and controls (CFR: 41.7)	4.1	56
510001 (SF8 CWS) Circulating Water								X				Ability to (a) predict the impacts of the following on the Circulating Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5, 41.1 / 43.5 / 45.3, 45.6, 45.13) A2.04 System leakage/rupture	3.2	60
K/A Category Point Totals:	1	1	1	1	1	1	1	1	1	1	1	Group Point Total:		11

Form 4.1-COMMON Common Examination Outline

Facility: Grand Gulf		Date of Exam:				
Generic Knowledge and Abilities—Tier 3 (RO/SRO)						
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, 10 CFR Part 55 (CFR: 41.10 / 43.2)	3.3	65		
	2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management (CFR: 41.1 / 41.5 / 41.10 / 43.6 / 45.6)	4.3	67		
	Subtotal		2			
2. Equipment Control	2.2.22	Knowledge of limiting conditions for operation and safety limits (CFR: 41.5 / 43.2 / 45.2)	4.0	68		
	2.2.43	Knowledge of the process used to track inoperable alarms (CFR: 41.10 / 43.5 / 45.13)	3.0	64		
	Subtotal		2			
3. Radiation Control	2.3.12	Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters (CFR: 41.12 / 43.4 / 45.9 / 45.10)	3.2	66		
	Subtotal		1			
4. Emergency Procedures/ Plan	2.4.35	Knowledge of non-licensed operator responsibilities during an emergency (CFR: 41.10 / 43.1/ 43.5 / 45.13)	3.8	69		
	Subtotal		1			
Tier 3 Point Total				**6**		

Facility: Grand Gulf		Date of Exam:		
Theory—Tier 4 (RO)				
Category	K/A #	Topic	RO	
			IR	#
Reactor Theory	K1.11	292004 Reactivity Coefficients K1.11 Describe the effect on the magnitude of void coefficient from changes in the following: Core void fraction	2.6	71
	K1.10	292005 Control Rods K1.10 State the purpose of flux shaping	3.3	73
	K1.24	292008 Reactor Operational Physics K1.24 Describe the parameters to be monitored and controlled during rod pattern exchanges	3.2	74
	Subtotal		N/A	
Thermodynamics	K1.14	293004 Thermodynamic Processes K1.14 Explain the condensing process	2.7	72
	K1.05	293006 Fluid Statics and Dynamics K1.05 Explain operational implications of fluid/water hammer	3.3	70
	K1.04	293008 Thermal-Hydraulics K1.04 Describe means by which boiling improves convection heat transfer	2.7	75
	Subtotal		N/A	
Tier 4 Point Total				**6**

Form 4.1-BWR Pressurized-Water Reactor Examination Outline

Facility: Grand Gulf Date of Exam:

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	Total	A2	G	Total
1. Emergency and Abnormal Plant Evolutions	1				N/A					N/A				4	3	7
	2													2	1	3
	Tier Totals													6	4	10
2. Plant Systems	1													3	2	5
	2													1 1	1	3
	Tier Totals													5	3	8
3. Generic Knowledge and Abilities Categories	CO			EC				RC		EM				CO / EC / RC / EM		7
														2 / 2 / 1 / 2		
4. Theory	Reactor Theory				Thermodynamics											

Notes: CO = Conduct of Operations; EC = Equipment Control; RC = Radiation Control; EM = Emergency Procedures/Plan

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** These systems/evolutions are only included as part of the sample (as applicable to the facility) when Revision 2 of the K/A catalog is used to develop the sample plan

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									
295003 (APE 3) Partial or Complete Loss of AC Power					X		Ability to determine and/or interpret the following as they apply to Partial or Complete Loss of AC Power: (CFR: 41.10 / 43.5 / 45.13) AA2.03 Battery status	3.9	79
295004 (APE 4) Partial or Total Loss of DC Power									
295005 (APE 5) Main Turbine Generator Trip									
295006 (APE 6) Scram					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	4.5	81
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) AA2.03 Reactor pressure	4.2	78
295018 (APE 18) Partial or Complete Loss of CCW									
295019 (APE 19) Partial or Complete Loss of Instrument Air						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	82
295021 (APE 21) Loss of Shutdown Cooling									
295023 (APE 23) Refueling Accidents									
295024 (EPE 1) High Drywell Pressure									
295025 (EPE 2) High Reactor Pressure					X		Ability to determine and/or interpret the following as they apply to High Reactor Pressure: (CFR: 41.10 / 43.5 / 45.13) EA2.01 Reactor pressure	4.3	77
295026 (EPE 3) Suppression Pool High Water Temperature						X	2.4.23 Knowledge of the bases for prioritizing emergency operating procedures implementation (CFR: 41.10 / 43.5 / 45.13)	4.4	80
295027 (EPE 4) High Containment Temperature (Mark III Containment Only)									
~~295028 (EPE 5) High Drywell Temperature (Mark I and Mark II only)~~									
295030 (EPE 7) Low Suppression Pool Water Level									
295031 (EPE 8) Reactor Low Water Level									
295037 (EPE 14) Scram Condition Present and Reactor Power Above APRM Downscale or Unknown									
295038 (EPE 15) High Offsite Radioactivity Release Rate									
600000 (APE 24) Plant Fire on Site									
700000 (APE 25) Generator Voltage and Electric Grid Disturbances						X	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	4.1	76
K/A Category Totals:	0	0	0	0	4	3	Group Point Total:		7

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					X		Ability to determine and/or interpret the following as they apply to Loss of Main Condenser Vacuum: (CFR: 41.10 / 43.5 / 45.13) AA2.07 Turbine limitations	3.4	84
295008 (APE 8) High Reactor Water Level									
295009 (APE 9) Low Reactor Water Level									
295010 (APE 10) High Drywell Pressure									
295011 (APE 11) High Containment Temperature (Mark III Containment only)									
295012 (APE 12) High Drywell Temperature									
295013 (APE 13) High Suppression Pool Water Temperature/ 5									
295014 (APE 14) Inadvertent Reactivity Addition									
295017 (APE 17) High Offsite Release Rate									
295020 (APE 20) Inadvertent Containment Isolation						X	2.4.51 Knowledge of emergency operating procedure exit conditions (e.g., emergency condition no longer exists or severe accident guideline entry is required) (CFR: 41.10 / 43.5 / 45.13)	4.0	83
295022 (APE 22) Loss of Control Rod Drive Pumps									
295029 (EPE 6) High Suppression Pool Water Level					X		Ability to determine and/or interpret the following as they apply to High Suppression Pool Water Level: (CFR: 41.10 / 43.5 / 45.13) EA2.03 Drywell/containment water level	3.6	85
295032 (EPE 9) High Secondary Containment Area Temperature									
295033 (EPE 10) High Secondary Containment Area Radiation Levels									
295034 (EPE 11) Secondary Containment Ventilation High Radiation / 9									
295035 (EPE 12) Secondary Containment High Differential Pressure									
295036 (EPE 13) Secondary Containment High Sump/Area Water Level									
500000 (EPE 16) High Containment Hydrogen Concentration									
K/A Category Point Totals:	0	0	0	0	2	1	Group Point Total:		3

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 HPCI) 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								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 operations: (CFR: 41.5 / 43.5 / 45.6) A2.01 Pump trip	3.8	90
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	2.4.6 Knowledge of emergency and abnormal operating procedures major action categories (CFR: 41.10 / 43.5 / 45.13)	4.7	89
217000 (SF2, SF4 RCIC) Reactor Core Isolation Cooling														
218000 (SF3 ADS) Automatic Depressurization											X	2.4.12 Knowledge of operating crew responsibilities during emergency and abnormal operations (CFR: 41.10 / 45.12)	4.3	87
223002 (SF5 PCIS) Primary Containment Isolation/Nuclear Steam Supply Shutoff														
239002 (SF3 SRV) Safety Relief Valves														
259002 (SF2 RWLCS) Reactor Water Level Control								X				Ability to (a) predict the impacts of the following on the Reactor Water Level Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.6) A2.03 Loss of reactor water level input	3.8	86
261000 (SF9 SGTS) Standby Gas Treatment														
262001 (SF6 AC) AC Electrical Distribution														
262002 (SF6 UPS) Uninterruptable Power Supply (AC/DC)								X				Ability to (a) predict the impacts of the following on the Uninterruptable Power Supply (AC/DC) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 43.5 / 45.6) A2.05 Loss of UPS	3.9	88

263000 (SF6 DC) DC Electrical Distribution														
264000 (SF6 EGE) Emergency Generators (Diesel/Jet)														
300000 (SF8 IA) Instrument Air														
400000 (SF8 CCW) Component Cooling Water														
510000 (SF4 SWS) Service Water														
K/A Category Point Totals:	0	0	0	0	0	0	0	3	0	0	2	Group Point Total:		5

Form 4.1-BWR	BWR Examination Outline Plant Systems—Tier 2/Group 2 (RO/SRO)	Page 6

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														
~~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														
226001 (SF5 RHR CSS) RHR/LPCI: Containment Spray Mode														
230000 (SF5 RHR SPS) RHR/LPCI: Torus/Suppression Pool Cooling Mode														
233000 (SF9 FPCCU) Fuel Pool Cooling/Cleanup														
234000 (SF8 FH) Fuel Handling Equipment				X								Knowledge of Fuel Handling System design features and/or interlocks that provide for the following: (CFR: 41.7) K4.02 Prevention of control rod movement during core alterations	3.8	92
239001 (SF3, SF4 MRSS) Main and Reheat Steam														
239003 (SF9 MSIVLC) Main Steam Isolation Valve Leakage Control														
241000 (SF3 RTPRS) Reactor/Turbine Pressure Regulating														
245000 (SF4 MTGEN) Main Turbine Generator/Auxiliary														
256000 (SF2 CDS) Condensate														
259001 (SF2 FWS) Feedwater														
268000 (SF9 RW) Radwaste														

271000 (SF9 OG) Offgas								X				Ability to (a) predict the impacts of the following on the Offgas System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: (CFR: 41.5 / 45.6 / 43.5 / 45.8) A2.06 Offgas system holdup volume explosion/ fire	3.3	91
272000 (SF7, SF9 RMS) Radiation Monitoring														
286000 (SF8 FPS) Fire Protection														
288000 (SF9 PVS) Plant Ventilation											X	2.1.32 Ability to explain and apply system precautions, limitations, notes, or cautions (CFR: 41.10 / 43.2 / 45.12)	4.0	93
290001 (SF5 SC) Secondary Containment														
290003 (SF9 CRV) Control Room Ventilation														
290002 (SF4 RVI) Reactor Vessel Internals														
510001 (SF8 CWS) Circulating Water														
K/A Category Point Totals:				1				1			1	Group Point Total:		3

Form 4.1-COMMON Common Examination Outline

Facility: Grand Gulf		Date of Exam:				
Generic Knowledge and Abilities—Tier 3 (RO/SRO)						
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.15	Knowledge of administrative requirements for temporary management direction, such as standing orders, night orders, or operations memoranda (CFR: 41.10 / 45.12)			3.4	99
	2.1.42	Knowledge of new and spent fuel movement procedures (SRO Only) (CFR: 43.7 / 45.13)			3.4	100
	Subtotal				2	
2. Equipment Control	2.2.25	Knowledge of the bases in technical specifications for limiting conditions for operation and safety limits (SRO Only) (CFR: 43.2)			4.2	98
	2.2.37	Ability to determine operability or availability of safety-related equipment (SRO Only) (CFR: 43.2 / 43.5 / 45.12)			4.6	97
	Subtotal				2	
3. Radiation Control	2.3.6	Ability to approve liquid or gaseous release permits (CFR: 41.13 / 43.4 / 45.10)			3.8	95
	Subtotal				1	
4. Emergency Procedures/ Plan	2.4.16	Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, or severe accident management guidelines (CFR: 41.10 / 43.5 / 45.13)			4.4	94
	2.4.40	Knowledge of SRO responsibilities in emergency plan implementing procedures (SRO Only) (CFR: 43.5 / 45.11)			4.5	96
	Subtotal				2	
Tier 3 Point Total						**7**

Form 4.1-1 Record of Rejected Knowledge and Abilities

Refer to Examination Standard (ES)-4.2, "Developing Written Examinations," Section B.3, for deviations from the approved written examination outline.

Tier/Group	Randomly Selected K/A	Reason for Rejection
1 / 1	Question #13 295018 Partial or Complete Loss of CCW AK2.05 - Knowledge of the relationship between Partial or Complete Loss of Component Cooling Water and the following systems or components: RHR/LPCI	An acceptable question could not be developed for the randomly sampled K/A due to lack of significant, relevant link between loss of CCW and RHR/LPCI. Randomly reselected K/A 295018 Partial or Complete Loss of CCW AK2.04 - Knowledge of the relationship between Partial or Complete Loss of Component Cooling Water and the following systems or components: Reactor recirculation system.
1 / 2	Question #21 50000 High Containment Hydrogen Concentration EK3.05 - Knowledge of the reasons for the following responses or actions as they apply to High Containment Hydrogen Concentration: Operation of wetwell (suppression pool) sprays	An acceptable question could not be developed for the randomly sampled K/A because the facility does not operate Suppression Pool spray in response to high Containment hydrogen concentration. Randomly reselected K/A 50000 High Containment Hydrogen Concentration EK3.10 - Knowledge of the reasons for the following responses or actions as they apply to High Containment Hydrogen Concentration: Operation of the hydrogen igniters (BWR 6).

2 / 1	Question #42 262001 AC Electrical Distribution K4.08 - Knowledge of AC Electrical Distribution design features and/or interlocks that provide for the following: Alternate breaker control methods	An acceptable question could not be developed for the randomly sampled K/A without being very similar to Question #12. Randomly reselected K/A 262001 AC Electrical Distribution K4.01 - Knowledge of AC Electrical Distribution design features and/or interlocks that provide for the following: Lockouts.
2 / 1	Question #48 510000 Service Water A2.02 - Ability to (a) predict the impacts of the following on the Service Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Abnormal valve positions	An acceptable question could not be developed for the randomly sampled K/A due to lack of relevant procedural guidance related to abnormal Service Water valve positions. Randomly reselected K/A 510000 Service Water A2.08 - Ability to (a) predict the impacts of the following on the Service Water System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Abnormal intake water level.
1 / 1	Question #78 295016 Control Room Abandonment AA2.05 - Ability to determine and/or interpret the following as they apply to Control Room Abandonment: Drywell pressure	An acceptable question could not be developed for the randomly sampled K/A due to lack of procedural tie between Control Room Abandonment and Drywell pressure. Randomly reselected K/A 295016 Control Room Abandonment AA2.03 - Ability to determine and/or interpret the following as they apply to Control Room Abandonment: Reactor pressure.

1 / 1	Question #80 295026 Suppression Pool High Water Temperature 2.4.2 - Knowledge of system setpoints, interlocks and automatic actions associated with emergency and abnormal operating procedure entry conditions	An acceptable question could not be developed for the randomly sampled K/A at the SRO license level because systems knowledge related to EP and ONEP entry conditions is RO level knowledge. Randomly reselected K/A 295026 Suppression Pool High Water Temperature 2.4.23 - Knowledge of the bases for prioritizing emergency operating procedures implementation.
2 / 1	Question #86 259002 Reactor Water Level Control A2.07 - Ability to (a) predict the impacts of the following on the Reactor Water Level Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of comparator bias signal	An acceptable question could not be developed for the randomly sampled K/A due to lack of relevant signal at facility. Randomly reselected K/A 259002 Reactor Water Level Control A2.03 - Ability to (a) predict the impacts of the following on the Reactor Water Level Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of reactor water level input.

2 / 2	Question #92 234000 Fuel Handling Equipment K4.06 - Knowledge of Fuel Handling System design features and/or interlocks that provide for the following: Protection from dropping a fuel assembly	An acceptable question could not be developed for the randomly sampled K/A without being relatively simplistic and/or not at the SRO level. Randomly reselected K/A 234000 Fuel Handling Equipment K4.02 - Knowledge of Fuel Handling System design features and/or interlocks that provide for the following: Prevention of control rod movement during core alterations.
1 / 1	Question #76 700000 Generator Voltage and Electric Grid Disturbances 2.4.18 - Knowledge of the specific bases for emergency and abnormal operating procedures	An acceptable question could not be developed for the randomly sampled generic K/A at the SRO level due to limited specific bases in the abnormal operating procedure for Generator Voltage and Electric Grid Disturbances. Randomly reselected K/A 700000 Generator Voltage and Electric Grid Disturbances 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.

Form 3.2-1 Administrative Topics Outline

Facility: Grand Gulf	Date of Examination: Jan 2024
Examination Level: RO	Operating Test Number: 2024

Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)	Type Code (Step 3)
Conduct of Operations	Loss of Shutdown Cooling, Time to 200°F Determination K/A 2.1.25 (3.9), 05-1-02-III-1	P, D, R **2021 NRC**
Conduct of Operations	Perform Daily Operating Logs K/A 2.1.18 (3.6), 06-OP-1000-D-0001	N, S
Equipment Control	Determine Isolation Boundary for SBGT Fan Work K/A 2.2.13 (4.1), EN-OP-102	N, R
Radiation Control		
Emergency Plan	Reactor Water Level Determination K/A 2.4.34 (4.2), 05-1-02-II-1	D, R

Location:

(C)ontrol room, (S)imulator, or Class(R)oom

Source and Source Criteria:

(P)revious two NRC exams (no more than one JPM that is **randomly selected** from last two NRC exams)
(D)irect from bank (no more than three for ROs, no more than four for SROs and RO retakes)
(N)ew or Significantly (M)odified from bank (no fewer than one)

Form 3.2-1 Administrative Topics Outline

Facility: Grand Gulf	Date of Examination: Jan 2024
Examination Level: SRO	Operating Test Number: 2024

Administrative Topic (Step 1)	Activity and Associated K/A (Step 2)	Type Code (Step 3)
Conduct of Operations	Reactor Water Chemistry Required Actions K/A 2.1.34 (3.5), 05-1-02-V-12	P, D, R **2021 NRC**
Conduct of Operations	Determine Reportability Requirements K/A 2.1.18 (3.8), EN-LI-108	M, R
Equipment Control	Review Daily Operating Logs, Determine Technical Specification Impact K/A 2.2.12 (4.1), 06-OP-1000-D-0001, Technical Specifications	N, R
Radiation Control	Emergency Exposure Limits K/A 2.3.12 (3.7), 10-S-01-17, EN-RP-201	D, R
Emergency Plan	Perform Emergency Classification K/A 2.4.41 (4.6), 10-S-01-1	D, R

Location:

(C)ontrol room, (S)imulator, or Class(R)oom

Source and Source Criteria:

(P)revious two NRC exams (no more than one JPM that is **randomly selected** from last two NRC exams)
(D)irect from bank (no more than three for ROs, no more than four for SROs and RO retakes)
(N)ew or Significantly (M)odified from bank (no fewer than one)

Form 3.2-2 Control Room/In-Plant Systems Outline

Facility: Grand Gulf	Date of Examination: Jan 2024 Operating Test Number: 2024
Exam Level: RO / SRO-I / **SRO-U**	

System/JPM Title	Type Code	Safety Function
Control Room Systems		
S1. Rotate CRD Pumps; CRD Pump Low Oil Pressure K/A 201001 A4.01 (3.7), 04-1-01-C11-1, 04-1-02-1H13-P601-22A-G3	P, D, A, S 2021 NRC	1
C1. Defeat Feedwater Pump Level 9 Trips K/A 259001 A3.10 (3.9), 05-S-01-EP-1 Attachment 6	D, C	2
S2. Open MSIVs; Steam Leak Develops with Failure to Automatically Isolate **K/A 239001 A4.01 (4.4), 04-1-01-B21-1, ARI P601-18A-A3**	**M, A, L, S**	**3**
S3. Perform LPCS Surveillance, High Pump Current **K/A 209001 A4.01 (4.3), 06-OP-1E21-Q-0006, ARI P601-21A-A7**	**N, A, EN, S**	**4**
S4. Perform Containment Venting K/A 223001 A2.06 (4.0/4.1), 05-S-01-EP-1 Attachment 14	D, L, S	5
S5. Withdraw SRMs **K/A 215004 A4.04 (3.5), 04-1-01-C51-1**	**N, L, S**	**7**
S6. Swap Plant Air Compressors K/A 300000 A4.02 (3.5), 04-1-01-P51-1	N, S	8
S7. Perform ARM Functional Test (RO Only) K/A 272000 A4.02 (3.2), 04-1-03-D21-1	D, S	9

In-Plant Systems		
P1. Hydrogen Igniter Alternate Power **K/A 223001 A4.14 (3.5), 05-S-01-STRATEGY**	**D, E, R**	**5**
P2. Battery Charger Startup with Low Current **K/A 263000 A2.03 (3.4/3.2), 04-1-01-L11-1**	**M, A**	**6**
P3. Initiate Fire Suppression to Standby Gas Treatment Filter Train K/A 286000 2.1.30 (4.4/4.0), 04-1-02-1H13-P870-2A-B2	D, E, R	8

Code	License Level Criteria		
	RO	SRO-I	SRO-U
(A)lternate path	4–6 **(4)**	4–6 **(4)**	2–3 **(3)**
(C)ontrol room	**(1)**	**(1)**	**(0)**
(D)irect from bank	≤ 9 **(6)**	≤ 8 **(5)**	≤ 4 **(1)**
(E)mergency or abnormal in-plant	≥ 1 **(2)**	≥ 1 **(2)**	≥ 1 **(1)**
(EN)gineered safety feature (for control room system)	≥ 1 **(1)**	≥ 1 **(1)**	≥ 1 **(1)**
(L)ow power/shutdown	≥ 1 **(3)**	≥ 1 **(3)**	≥ 1 **(2)**
(N)ew or (M)odified from bank (must apply to at least one alternate path JPM)	≥ 2 **(5/3)**	≥ 2 **(5/3)**	≥ 1 **(4/3)**
(P)revious two exams (randomly selected)	≤ 3 **(1)**	≤ 3 **(1)**	≤ 2 **(0)**
(R)adiologically controlled area	≥ 1 **(2)**	≥ 1 **(2)**	≥ 1 **(1)**
(S)imulator	**(7)**	**(6)**	**(3)**

Facility:	Grand Gulf	Scenario #:	NRC-1
Scenario Source:	New	Op. Test #:	2024
Examiners:		Applicants/ Operators:	
Initial Conditions:	The plant is operating at approximately 85% power. CRD pump B is tagged out of service for maintenance.		
Turnover:	Start EHC control fluid pump B and place EHC control fluid pump A in standby per 04-1-01-N32-1 Section 5.3.5. The procedure is in progress up to step 5.3.5.2. Then, withdraw control rods to raise the rod line per the provided reactivity instructions.		
Critical Tasks:	CT-1: Given the plant operating at power with rising Drywell pressure and a failure of RPS to scram the Reactor, the crew will insert control rods by manually initiating ARI, in accordance with 05-1-02-I-1 and/or EP-2A. ARI must be manually initiated within 5 minutes of when Drywell pressure exceeds 1.23 psig. CT-2: Given a coolant leak, a loss of high-pressure injection systems, and the inability to restore and maintain Reactor water level above -191”, the crew will perform an Emergency Depressurization, in accordance with EP-2, before Reactor water level reaches -191”.		

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Swap EHC Pumps SOI 04-1-01-N32-1
2	N/A	R – ATC, SRO	Withdraw Control Rods SOI 04-1-01-C11-2
3	c51009_2 c51010_3	I – ATC, SRO MC – ATC TS – SRO	Two APRMs Fail Sequentially SOI 04-1-01-C51-1, Technical Specification 3.3.1.1
4	g33033	C – BOP, SRO MC – BOP TS – SRO	RWCU Leak into Containment, RWCU Fails to Automatically Isolate 05-1-02-III-5, 04-1-01-G33-1, Technical Specification 3.3.6.1
5	fw123b rr217a	C – All MC – ATC	Feedwater Pump Trip, Recirc FCV A Fails to Runback ONEP 05-1-02-V-7, ONEP 05-1-02-III-3
6	rr063	M – All	Coolant Leak in Drywell ONEP 05-1-02-I-1, EP-2, EP-3
7	c71162	I – ATC, SRO MC – ATC	Automatic and Manual RPS Scrams Fail, ARI Works ONEP 05-1-02-I-1, EP-2, EP-2A
8	fw115	C – All	Condensate Pumps Trip EP-2
9	e22052	C – All MC – ATC, BOP	HPCS Pump Trip EP-2
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control			

GGNS 01-2024 NRC Scenario 1

Initial Conditions: -

- Plant is operating at approximately 85% power, middle of cycle

Inoperable Equipment:

- CRD pump B is tagged out of service for maintenance.

Planned activities:

- Start EHC control fluid pump B and place EHC control fluid pump A in standby per 04-1-01-N32-1 Section 5.3.5. The procedure is in progress up to step 5.3.5.2.
- Then, withdraw control rods to raise the rod line per the provided reactivity instructions.

Scenario Notes:

- Validation Time: 60 minutes

SCENARIO ACTIVITIES:

Event 1 – Swap EHC Pumps (Normal evolution)

After the crew assumes the shift, the BOP will start EHC control fluid pump B and place EHC control fluid pump A in standby per 04-1-01-N32-1 Section 5.3.5. The procedure is in progress up to step 5.3.5.2.

Event 2 – Withdraw Control Rods (Reactivity evolution)

Following the normal evolution, the ATC will withdraw control rods to adjust the rod pattern. Control rods 24-33, 32-41, 40-33, and 32-25 will be withdrawn from position 08 to position 12.

Event 3 – Two APRMs Fail Sequentially (Triggered by Lead Examiner)

APRM channel 2 will fail upscale. The crew will bypass APRM channel 2. Then, APRM 3 will fail downscale. The SRO will determine the Technical Specification impact.

Tech Specs should be referred to and the following LCOs identified:		
3.3.1.1	Condition A, One or more required channels inoperable	
	A.1	Place channel in trip (12 hours)
	A.2	Place associated trip system in trip (12 hours)
TRM 3.3.2.1	Condition A, One or more required channels nonfunctional	
	A.1	Enter the Condition referenced in Table TR3.3.2.1-2 for that channel (immediately)
	Condition B, As required by Required Action A.1 and referenced in Table TR3.3.2.1-2	
	B.1	Enter Condition C (7 days)

Event 4 – RWCU Leak into Drywell, RWCU Fails to Automatically Isolate (Triggered by Lead Examiner)

A leak will develop from RWCU into the heat exchanger room inside Containment. Room temperature will rise, but RWCU will fail to automatically isolate. The crew will execute 05-1-02-III-5, Automatic Isolations ONEP. The crew will manually isolate RWCU to terminate the leak. The SRO will determine the Technical Specification impact.

Tech Specs should be referred to and the following LCOs identified:		
3.3.6.1	Condition B, One or more automatic Functions with isolation capability not maintained.	
	B.1	Restore isolation capability (1 hour)

Event 5 – Feedwater Pump Trip, Recirc FCV A Fails to Runback (Triggered by Lead Examiner)

Annunciator P680-2A-A12, RFPT B TRIP will alarm and reactor feed pump "B" will trip. With the recirc pumps operating in fast speed combined with only 1 RFPT in service and RPV level lowering to < 32"NR, a Recirc flow control runback should occur to lower power to within the capability of 1 RFPT (≈ 20% valve position). The recirc FCV runback will occur on the B loop only. The ATC should observe the A FCV did not respond and lower A loop flow to within TS limit for loop mismatch. If no action is taken, RPV level will lower to ≈ 20" NR and will then start slowly trending up.

05-1-02-V-7, Feedwater System Malfunctions ONEP should be entered that ensures the runback has occurred.

05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP should be entered to ensure:

- Recirc loop flow mismatch is within limits
- Perform plot on power/flow map
- OPRMs are armed
- THI watch established

The A RFPT will speed up to maintain RPV level and will result in receipt of alarm P680-2A-E2, RFP A VIBR HI. An NLO should be dispatched to investigate. When dispatched, the booth operator will inform the control room the vibration levels are below alarm values and will reset the alarm.

Event 6 – Coolant Leak in Drywell (Triggered by Lead Examiner)

A small coolant leak will develop from a Recirc loop into the Drywell. Drywell pressure and temperature will begin to rise. The crew will enter 05-1-02-I-1, Reactor Scram (see event 7). The crew will enter EP-2, RPV Control, and EP-3, Primary Containment Control. Initially, the crew will control Reactor water level with Condensate/Feedwater. The coolant leak will worsen later worsen and injection systems will be lost (events 8 and 9).

Event 7 – Automatic and Manual RPS Scrams Fail, ARI Works (Initial Setup)

When either the ATC attempts a manual Reactor scram or Drywell pressure exceeds the scram setpoint, RPS will fail to actuate and insert control rods. The ATC will manually initiate ARI **(critical task)** and all control rods will insert.

Event 8 – Condensate Pumps Trip (Triggered by Lead Examiner)

Once the Reactor is stabilized after the scram, Condensate pumps will trip. This will result in a loss of high-pressure Feedwater injection. The crew will transition to utilizing HPCS to maintain Reactor water level.

Event 9 – HPCS Pump Trip (Triggered by Lead Examiner)

Approximately two minutes after the loss of Condensate pumps, the HPCS pump will also trip. The crew will inject with the only remaining high-pressure injection sources, CRD and SLC, but this will not be enough injection to maintain Reactor water level. The crew will determine that Reactor water level cannot be maintained above -166” and perform an Emergency Depressurization **(critical task)**. The crew will then utilize low-pressure systems to control Reactor water level above top of active fuel.

Once the Reactor has been Emergency Depressurized, and Reactor water level is controlled above top of active fuel, this scenario can be terminated.

GGNS 01-2024 NRC Scenario 2

Facility:	Grand Gulf	Scenario #:	NRC-2
Scenario Source:	New	Op. Test #:	2024
Examiners:		Applicants/	
		Operators:	
Initial Conditions:	The plant is operating at approximately 100% power. CCW pump C is out of service for maintenance. Suppression Pool level is low.		
Turnover:	Makeup to the Suppression Pool using HPCS per 04-1-01-E22-1. Then, lower Reactor power to 1400 MWe using Recirc flow per the provided reactivity instructions.		
Critical Tasks:	CT-1: Given a failure to scram with Reactor power above 5%, the crew will lower Reactor power by one or more of the following methods, in accordance with EOP-2A: • Terminating and preventing all RPV injection except SLC, RCIC and CRD • Tripping Recirc pumps The Reactor power reduction must be initiated within 90 seconds of the start of the failure to scram. CT-2: Given a failure to scram, the crew will initiate control rod insertion, in accordance with EOP-2A. All insertable control rods must be inserted within one hour of the start of the failure to scram. CT-3: Restore instrument air to the containment and drywell (to maintain MSIVs open) prior to SRVs cycling to control RPV pressure.		

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Makeup to the Suppression Pool from HPCS SOI 04-1-01-E22-1
2	N/A	R – ATC, SRO	Lower Reactor Power using Recirc IOI 03-1-01-2
3	p43152a	C – BOP, SRO MC – BOP	TBCW Pump Trip, Standby Pump Fails to Automatically Start EN-OP-200
4	e22054 e22054	I – BOP, SRO MC – BOP TS – SRO	DW Press instruments B21-PIS-N667C &G fail upscale causing Initiation of Div III ECCS, HPCS Initiates 04-1-01-E22-1 Attachment V, Technical Specifications 3.5.1, 3.3.5.1, 3.3.6.1, 3.3.8.1
5	rr012	C – ATC, SRO TS – SRO	Recirc Pump Trip ONEP 05-1-02-III-3, Technical Specifications 3.4.1, 3.3.1.1
6	rr071	M – All	Fuel Failure ONEP 05-1-02-II-2, ONEP 05-1-02-I-1, EP-2
7	c11167	C – All MC – ATC, BOP	RPS and ARI Fail to Insert Control Rods EP-2A
8	c41c001	C – BOP, SRO	SLC Pumps Trip EP-2A

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control

GGNS 01-2024 NRC Scenario 2

Initial Conditions:

- Plant is operating at approximately 100% power, Middle of Cycle
- Suppression Pool level is low.

Inoperable Equipment:

- CCW pump C is tagged out of service for maintenance.

Planned activities:

- Makeup to the Suppression Pool using HPCS per 04-1-01-E22-1. Raise Suppression Pool level to 18.5 feet.
- Then, lower Reactor power to 1400 MWe using Recirc flow per the provided reactivity instructions.

Scenario Notes:

- Validation Time: 60 minutes

<u>SCENARIO ACTIVITIES:</u>

<u>Event 1 – Makeup to Suppression Pool Makeup using HPCS</u> (Normal evolution)

After the crew assumes the shift, the BOP will makeup to the Suppression Pool makeup using HPCS per 04-1-01-E22-1, HPCS SOI.

<u>Event 2 – Lower Reactor Power using Recirc</u> (Reactivity evolution)

Following the normal evolution, the ATC will lower Reactor power to approximately 1400 MWe using Recirc flow.

<u>Event 3 – TBCW Pump Trip, Standby Pump Fails to Automatically Start</u> (Triggered by Lead Examiner)

A TBCW pump trip will occur and the standby pump will fail to auto start. The BOP operator should recognize the failure of the standby pump to start and start the standby pump.

<u>Event 4 – Spurious Initiation of Div 3 ECCS, HPCS initiates</u> (Triggered by Lead Examiner)

An upscale failure of DW Press instruments B21-PIS-N667 C &G causes initiation of Div III ECCS, HPCS initiates. The crew uses the hardcard 04-1-01-E22-1 Attachment V to secure HPCS, the failed drywell pressure instruments and HPCS overridden off results in entry into multiple LCOs. The SRO will determine the Technical Specification impact.

Tech Specs should be referred to and the following LCOs identified:		
3.5.1	Condition B, HPCS inoperable	
	B.1	Verify by administrative means RCIC is operable (1 hour)
	B.2	Restore HPCS to operable (14 days)
3.3.5.1	Condition A, One or more required channels inoperable	
	A.1	Enter the Condition referenced in Table 3.3.5.1-1 for the channel (Immediately)
	Condition B,	
	B.2	Declare HPCS INOP (1 hour)
3.3.6.1	Condition A, One or more required channels inoperable	
	A.1	Place channel in trip (24 hours)
	Condition F, as required by required action C .1	
	F.1	Isolate the affected penetration flow path. (1 hour)

Tech Specs should be referred to and the following LCOs identified:		
3.5.1	Condition B, HPCS inoperable	
	B.1	Verify by administrative means RCIC is operable (1 hour)
	B.2	Restore HPCS to operable (14 days)
3.3.5.1	Condition A, One or more required channels inoperable	
	A.1	Enter the Condition referenced in Table 3.3.5.1-1 for the channel (Immediately)
	Condition B,	
	B.2	Declare HPCS INOP (1 hour)
3.3.6.1	Condition A, One or more required channels inoperable	
	A.1	Place channel in trip (24 hours)
	Condition F, as required by required action C .1	
	F.1	Isolate the affected penetration flow path. (1 hour)

<u>Event 5 – Recirc Pump Trip</u> (Triggered by Lead Examiner)

A Recirc pump trip occurs. The CRS will enter 05-1-02-III-3, Reduction in Recirculation System Flowrate ONEP. The ATC should perform a plot on the Power / Flow operating map, verify OPRMs are armed, and assume THI monitoring. Control rod insertion will be required to gain entry to within the restraints of Attachment 3 of the ONEP. CRS will determine the Technical Specification impact.

Tech Specs should be referred to and the following LCOs identified:	
3.4.1	Establish single loop limits (12 hours)
3.3.1.1	Establish single loop limits (12 hours)
TRM should be referred to and the following identified:	
3.3.2.1	Establish single loop limits (12 hours)

<u>Event 6 – Fuel Failure</u> (Triggered by Lead Examiner)

Fuel failure will occur. High radiation alarms will be received for Drywell airborne, Offgas Pretreatment, and Main Steam Line radiation. The CRS should enter 05-1-02-II-2, Offgas Activity High ONEP. The CRS should determine that without action, Offgas Pre-treat radiation values will exceed 1400 mr/hr and direct core flow to be lowered to 70 mlbm/hr. With the lowering of core flow, 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP should be entered. Offgas radiation levels will continue to worsen and the crew will attempt a manual Reactor scram.

<u>Event 7 – RPS and ARI Fail to Insert Control Rods</u> (Initial Setup)

When the crew attempts the manual Reactor scram, RPS and ARI will fail to insert control rods. The CRS will enter EP-2, RPV Control, and transition to EP-2A, ATWS RPV Control. The crew will take actions to lower Reactor power, including lowering Recirc flow, tripping Recirc pumps, intentionally lowering Reactor water level, and/or initiating boron injection with SLC **(critical task)**. The crew will also take action to insert control rods by venting the scram air header, tripping RPS circuit breakers, and/or manually inserting control rods **(critical task)**.

<u>Event 8 – SLC Pumps Trip</u> (Initial Setup)

When the crew starts SLC pumps, they will trip after one minute of operation. This will limit the negative reactivity from boron injection and lead to a higher power level for the remainder of the ATWS.

Once control rods are inserted and Reactor water level is controlled above top of active fuel, this scenario can be terminated.

Termination:

- When directed by Lead Evaluator, the booth operator will perform the following:
  - 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.

GGNS 01-2024 NRC Scenario 3

Facility:	Grand Gulf	Scenario #:	NRC-3
Scenario Source:	New	Op. Test #:	2024
Examiners:		Applicants/	
		Operators:	

Initial Conditions:	The plant is operating at approximately 4% power during startup.
Turnover:	IRM B is bypassed. Return IRM B to service per 04-1-01-C51-1 section 5.3. Then, continue the startup by withdrawing control rods per the provided reactivity instructions. Step 68 is complete. Start at step 69. Raise power in preparation for transitioning to Mode 1.
Critical Tasks:	CT-1: Given a leak from the Suppression Pool and lowering Suppression Pool water level, isolate the leak, in accordance with EP-4. The leak must be isolated in time to prevent Suppression Pool water level from lowering below 14.5'. CT-2: Given the plant operating at power with an un-isolable primary system discharging into Secondary Containment, the crew will insert a manual Reactor scram, in accordance with EP-4. The scram must be inserted before two areas reach a Maximum Safe Temperature. CT-3: Given an un-isolable primary system discharging into Secondary Containment and two areas exceeding Maximum Safe Temperatures, the crew will perform an Emergency Depressurization, in accordance with EP-4. The Emergency Depressurization must be initiated within 15 minutes of when the second Maximum Safe Temperature is exceeded.

GGNS 01-2024 NRC Scenario 3

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Return IRM to Service 04-1-01-C51-1
2	N/A	R – ATC, SRO	Withdraw Control Rods 04-1-01-C11-2
3	z021021	C – ATC, SRO TS – SRO	Control Rod Drift In ONEP 05-1-02-IV-1, Technical Specification 3.1.3
4	e12278C ct219c	C – BOP, SRO TS – SRO	Suppression Pool Leak from RHR, RHR Room Door Fails EP-3, EP-4, Technical Specification 3.5.1
5	fw274	I – ATC, SRO MC – ATC	Startup Level Control Valve Drifts Closed in Auto ONEP 05-1-02-V-7
6	e51050 e51187a e51187b ms066a	M – All	RCIC Steam Leak in Auxiliary Building, RCIC Isolation Valve Breaker Trip on Stroke. >Max Safe in 2 areas. EP-4, ONEP 05-1-02-I-1, EP-2, EP-2 ED Leg
7	e51319 e51320	C – BOP, SRO MC – BOP	RCIC Fails to Automatically Isolate EP-4
8	tc248	C – ATC, SRO MC – ATC	EHC Pumps Trip EP-2
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (TS)Tech Spec, (MC)Manual Control			

Initial

Conditions:

- Plant is operating at approximately 4% power during a startup

Inoperable Equipment:

- IRM B is bypassed

Planned activities:

- Return IRM B to service per 04-1-01-C51-1 section 5.3.
- Then, continue the startup by withdrawing control rods per the provided reactivity instructions. Step 68 is complete. Start at step 69. Raise power in preparation for transitioning to Mode 1.

Scenario Notes:

- Validation Time: 60 minutes

SCENARIO ACTIVITIES:

Event 1 – Return IRM to Service (Normal evolution)

After the crew assumes the shift, the BOP will return IRM B to service per 04-1-01-C51-1 section 5.3.

Event 2 – Withdraw Control Rods (Reactivity evolution)

Following the normal evolution, the ATC will withdraw control rods to continue the plant startup.

Event 3 – Control Rod Drift In (Triggered by Lead Examiner)

Control rod 56-53 will drift in. The crew will execute 05-1-02-IV-1, Control Rod/Drive Malfunctions ONEP. Per the ONEP, direction should be given to fully insert 56-53 by using the IN TIMER SKIP pushbutton. and to maintain an insert signal until the HCU can be isolated by closing C11-103 and C11-105 located inside containment. The CRS will determine the Technical Specification impact.

Tech Specs should be referred to and the following LCO identified:		
3.1.3	Condition C, One or more control rods inoperable for reasons other than Conditions A or B	
	C.1	Fully insert inoperable control rod (3 hours)
	C.2	Disarm the associated CRD (4 hours)

Event 4 – Suppression Pool Leak from RHR, RHR Room Door Fails (Triggered by Lead Examiner)

A leak will develop from RHR C piping. Additionally, the watertight door for the associated room will fail. This will cause Suppression Pool water level to lower. 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. The crew will isolate the leak by closing RHR C suction valve E12-F004C **(critical task)**. RHR C jockey pump will trip due to submergence (supplies keepfill for suppression pool level instrumentation reference legs.)The crew may initiate Suppression Pool Makeup to restore Suppression Pool level, as time permits. The CRS will determine the Technical Specification impact.

Tech Specs should be referred to and the following LCOs identified:		
3.5.1	Condition A, One low pressure ECCS injection/spray subsystem inoperable	
	A.1	Restore low pressure ECCS injection/spray subsystem to operable status (7 days)
3.6.2.2	Condition A, Suppression pool water level not within limits	
	A.1	Restore suppression pool water level to within limits (2 hours)
3.3.6.4	SPMU System Instrumentation (Function3)	
	C.1	Declare SPMU subsystem INOP (1 hour)
	C.2	Restore channel to operable status (24 hours)

Event 5 – Startup Level Control Valve Drifts Closed in Auto (Triggered by Lead Examiner)

The startup level controller, C34-R602 (P680-1B), output will start lowering and Reactor level will slowly lower, resulting in receipt of alarm P680-3A-A3, RX LVL 40"/32" HI/LO. The crew will execute 05-1-02-V-7, Feedwater System Malfunctions ONEP. The ATC will take manual control of the controller and adjust to maintain Reactor level.

Event 6 – RCIC Steam Leak in Auxiliary Building, RCIC Isolation Valve Breaker Trip on Stroke (Triggered by Lead Examiner)

A steam leak will develop from RCIC into the Auxiliary Building. The leak will be unisolable due to trip of breakers to isolation valves E51-F063 and E51-F064 after manual isolation is attempted. The crew will execute EP-4 due to high area temperatures. With an unisolable primary system discharging into the Auxiliary Building and area temperatures approaching/exceeding a Max Safe value, the crew will enter EP-2 and scram the Reactor **(critical task)**. Area temperatures will continue to degrade. When a 2nd area exceeds the Max Safe temperature, the crew will perform an Emergency Depressurization **(critical task)**.

<u>Event 7 – RCIC Fails to Automatically Isolate</u> (Initial Setup)

RCIC will fail to automatically isolate on high temperature. The crew will attempt to manually isolate RCIC. The breakers for isolation valves E51-F063 and E51-F064 will trip while the valves are stroking, resulting in the inability to fully isolate the steam leak from RCIC.

<u>Event 8 – EHC Pumps Trip</u> (Automatic)

When Reactor pressure lowers below approximately 830 psig, the running EHC pump will trip. Additionally, the standby EHC pump will receive a trip signal. This will result in a loss of ability to use Turbine Bypass Valves to control Reactor pressure. This will limit the crew's ability to lower Reactor pressure prior to Emergency Depressurization. The crew will use alternate systems (SRVs, RCIC) to control Reactor pressure, as required.

<u>Termination:</u>

- When directed by Lead Evaluator, the booth operator will perform the following:
  - 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.