Facility: **Grand Gulf Nuclear Station** Date of Exam: **2/10/2020**

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	4	4	3	N/A			3	3	N/A		3	**20**	4	3	7
	2	1	1	1				1	1			2	**7**	1	2	3
	Tier Totals	**5**	**5**	**4**				**4**	**4**			**5**	**27**	5	5	10
2. Plant Systems	1	3	3	2	2	2	2	2	2	2	3	3	**26**	3	2	5
	2	1	1	1	2	1	1	1	1	1	1	1	**12**		2	1	3
	Tier Totals	**4**	**4**	**3**	**4**	**3**	**3**	**3**	**3**	**3**	**4**	**4**	**38**	5	3	8

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

Note:
1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it is replaced by a K/A from another Tier 3 category.)
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points, and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the outline. Systems or evolutions that do not apply at the facility should be deleted with justification. Operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
4. Select topics from as many systems and evolutions as possible. Sample every system or evolution in the group before selecting a second topic for any system or evolution.
5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher. Use the RO and SRO ratings for the RO and SRO-only portions, respectively.
6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' IRs for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. If fuel-handling equipment is sampled in a category other than Category A2 or G* on the SRO-only exam, enter it on the left side of Column A2 for Tier 2, Group 2. (Note 1 does not apply.) Use duplicate pages for RO and SRO-only exams.
9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

G* Generic K/As

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

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

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4						✓	2.2.22 Knowledge of limiting conditions for operations and safety limits.	4.0	1
295003 Partial or Complete Loss of AC / 6			✓				AK3.05 Knowledge of the reasons for the following responses or actions as they apply to partial or complete loss of AC power: AK3.05 Reactor SCRAM	3.7	2
295004 Partial or Total Loss of DC Pwr / 6				✓			AA1.03 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : A.C. electrical distribution	3.4	3
295005 Main Turbine Generator Trip / 3					✓		AA2.04 Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : Reactor Pressure	3.7	4
295006 SCRAM / 1	✓						AK1.02 Knowledge of the operational implications of the following concepts as they apply to SCRAM : Shutdown Margin	3.4	5
295016 Control Room Abandonment / 7		✓					AK2.02 Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Local control stations	4.0	6
295018 Partial or Total Loss of CCW / 8					✓		AA2.02Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Cooling Water Temperatures	3.1	7
295019 Partial or Total Loss of Inst. Air / 8		✓					AK2.09 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Containment	3.3	8
295021 Loss of Shutdown Cooling / 4			✓				AK3.01 Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING : Raising reactor water level	3.3	9
295023 Refueling Acc / 8						✓	2.4.41 Knowledge of the emergency action level thresholds and classifications	2.9	10
295024 High Drywell Pressure / 5				✓			EA1.05 Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: RPS	3.9	11
295025 High Reactor Pressure / 3	✓						EK1.03 Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : Safety/relief valve tailpipe Temperature/pressure relationships	3.6	12
295026 Suppression Pool High Water Temp. / 5			✓				EK3.04 Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: SBLC injection	3.7	13
295027 High Containment Temperature / 5	✓						EK1.03 Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) : Containment integrity: Mark-III	3.8	14
295028 High Drywell Temperature / 5				✓			EK3.04 Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : Increased drywell cooling	3.6	15
295030 Low Suppression Pool Wtr Lvl / 5		✓					EK2.08 Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: SRV discharge submergence	3.5	16
295031 Reactor Low Water Level / 2						✓	2.4.3 Ability to identify post-accident instrumentation	3.7	17
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1					✓		EA2.03 Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : SBLC tank level	4.3	18

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - **Tier 1/Group 1 (RO)**						Form ES-401-1		
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295038 High Off-site Release Rate / 9		✓					EK2.10 Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Condenser air removal system	3.2	19
600000 Plant Fire On Site / 8									
700000 Generator Voltage and Electric Grid Disturbances / 6	✓						AK1.02 Knowledge of the operational implications of the | following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Over-excitation	3.3	20
K/A Category Totals:	**4**	**4**	**3**	**3**	**3**	**3**	**Group Point Total:**		**20**

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO)	Form ES-401-1	
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G*	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3									
295007 High Reactor Pressure / 3									
295008 High Reactor Water Level / 2				✓			AA1.06 Ability to operate and/or monitor the following as they apply to HIGH REACTOR WATER LEVEL : HPCS	2.8	21
295009 Low Reactor Water Level / 2									
295010 High Drywell Pressure / 5	✓						AK1.03 Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : Temperature Increases	3.2	22
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5									
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1									
295017 High Off-site Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7			✓				AK3.03 Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: Drywell/containment temperature response	3.2	23
295022 Loss of CRD Pumps / 1									
295029 High Suppression Pool Wtr Lvl / 5					✓		EA2.03 Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Drywell/containment water level	3.4	24
295032 High Secondary Containment Area Temperature / 5						✓	2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	4.2	25
295033 High Secondary Containment Area Radiation Levels / 9									
295034 Secondary Containment Ventilation High Radiation / 9									
295035 Secondary Containment High Differential Pressure / 5		✓					EK2.01 Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE and the following: Secondary containment ventilation	3.6	26
295036 Secondary Containment High Sump/Area Water Level / 5						✓	2.1.30 Ability to locate and operate components, including local controls.	4.4	27
500000 High CTMT Hydrogen Conc. / 5									
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	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode								✓				A2.03 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: valve Closures	3.2	28
205000 Shutdown Cooling			✓			✓						K3.01 Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor Pressure K6.08 Knowledge of the effect that a loss or malfunction of the following will have on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) : RHR Service Water	3.3 – 3.5	29 – 30
206000 HPCI														
20700 Isol Condenser														
209001 LPCS					✓						✓	K5.05Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : System Venting 2.2.40 Ability to apply Technical Specifications for a system.	2.5 — 3.4	31 — 32
209002 HPCS	✓				✓							K5.04 Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE CORE SPRAY SYSTEM (HPCS): Adequate core cooling K1.02 Knowledge of the physical connections and/or cause effect relationships between HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) and the following: Suppression Pool: BWR-5,6	3.8 — 3.5	33 — 53
211000 SLC							✓					A1.06 Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: Flow Indication	3.8	34
212000 RPS			✓							✓		K3.02 Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following: Primary containment isolation system/nuclear steam supply shut-off: Plant-Specific A4.08 Ability to manually operate and/or monitor in thecontrol room: Individual system relay status	3.7 — 3.4	35 — 36
215003 IRM		✓										K2.01 Knowledge of electrical power supplies to the following: IRM channels/detectors	2.5	37
215004 Source Range Monitor				✓								K4.04 Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Changing detector position	2.8	38

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	#
215005 APRM / LPRM		✓								✓		K2.02 Knowledge of electrical power supplies to the following: APRM Channels A4.04 Ability to manually operate and/or monitor in the control room: LPRM back panel switches, meters and indicating lights	2.6 — 3.2	39 — 40
217000 RCIC											✓	2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	4.2	41
218000 ADS							✓					A1.01 Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: ADS valve tail pipe temperatures	3.4	42
223002 PCIS/Nuclear Steam Supply Shutoff	✓											K1.14 Knowledge of the physical connections and/or causeeffect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: Containment drainage system	2.8	43
239002 SRVs				✓								K4.02 Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Minimizes containment fatigue duty cycles resulting from relief valve cycling during decay-heat-dominant period late in an isolation transient (LLS logic)	3.4	44
259002 Reactor Water Level Control									✓			A3.04 Ability to monitor automatic operations of the REACTOR WATER LEVEL CONTROL SYSTEM including: Changes in reactor feedwater flow	3.2	45
261000 SGTS	✓											K1.08 Knowledge of the physical connections and/or cause effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: Process radiation monitoring system	2.8	46
262001 AC Electrical Distribution											✓	2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	4.0	47
262002 UPS (AC/DC)						✓						K6.03 Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : Static inverter	2.7	48
263000 DC Electrical Distribution		✓										K2.01 Knowledge of electrical power supplies to the following: Major D.C. loads	3.1	49
264000 EDGs								✓				A2.01 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: Parallel operation of emergency generator	3.5	50

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	#
300000 Instrument Air										✓		A4.01 Ability to manually operate and / or monitor in the control room: Pressure Gauges	2.6	51
400000 Component Cooling Water									✓			A3.01 Ability to monitor automatic operations of the CCWS including: Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS	3.0	52
510000 (SF4 SWS*) Service Water (Normal and Emergency)														
K/A Category Point Totals:	**3**	**3**	**2**	**2**	**2**	**2**	**2**	**2**	**2**	**3**	**3**	**Group Point Total:**		**26**

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	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
201001 CRD Hydraulic				✓								K4.08 Knowledge of CONTROL ROD DRIVE HYDRAULIC SYSTEM design feature(s) and/or interlocks which provide for the following: Controlling control rod drive header pressure	3.1	54
201002 RMCS														
201003 Control Rod and Drive Mechanism														
201004 RSCS														
201005 RCIS			✓									K3.01 Knowledge of the effect that a loss or malfunction of the ROD CONTROL AND INFORMATION SYSTEM (RCIS) will have on following: Control rod drive system: BWR-6	3.3	57
201006 RWM														
202001 Recirculation	✓											K1.15 Knowledge of the physical connections and/or cause effect relationships between RECIRCULATION SYSTEM and the following: Nuclear boiler instrumentation (reactor water level/pressure)	3.2	55
202002 Recirculation Flow Control						✓						K6.06 Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM : Reactor water level	3.1	56
204000 RWCU														
214000 RPIS														
215001 Traversing In-Core Probe														
215002 RBM														
216000 Nuclear Boiler Inst.							✓					A1.02 Ability to predict and/or monitor changes in parameters associated with operating the NUCLEAR BOILER INSTRUMENTATION controls including: Removing or returning a sensor (transmitter) to service	2.9	58
219000 RHR/LPCI: Torus/Pool Cooling Mode											✓	2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.6	65
223001 Primary CTMT and Aux.														
226001 RHR/LPCI: CTMT Spray Mode		✓										K2.02 Knowledge of electrical power supplies to the following: Pumps	2.9	59
230000 RHR/LPCI: Torus/Pool Spray Mode														
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment														
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator														

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	#
245000 Main Turbine Gen. / Aux.										✓		A4.05 Ability to manually operate and/or monitor in the control room: Generator megawatt output	2.7	60
256000 Reactor Condensate				✓								K4.08 Knowledge of REACTOR CONDENSATE SYSTEM design feature(s) and/or interlocks which provide for the following: Dedicated ECCS water supply	3.6	61
259001 Reactor Feedwater					✓							K5.03 Knowledge of the operational implications of the following concepts as they apply to REACTOR FEEDWATER SYSTEM : Turbine operation: TDRFP's-Only	2.8	62
268000 Radwaste														
271000 Offgas														
272000 Radiation Monitoring									✓			A3.06 Ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including: Ventilation system isolation indications	3.4	63
286000 Fire Protection														
288000 Plant Ventilation														
290001 Secondary CTMT														
290003 Control Room HVAC														
290002 Reactor Vessel Internals								✓				A2.03 Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Control rod drop accident	3.6	64
51001 (SF8 CWS*) Circulating Water														
K/A Category Point Totals:	**1**	**1**	**1**	**2**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**Group Point Total:**		**12**

Facility: **Grand Gulf Nuclear Station** Date of Exam: **2/10/2020**

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1.8	Ability to coordinate personnel activities outside the control room.	3.4	66		
	2.1.19	Ability to use plant computers to evaluate system or component status.	3.9	67		
	Subtotal			**2**		
2. Equipment Control	2.2.40	Ability to apply Technical Specifications for a system.	3.4	68		
	2.2.43	Knowledge of the process used to track inoperable alarms.	3.0	69		
	2.2.44	Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	70		
	Subtotal			**3**		
3. Radiation Control	2.3.11	Ability to control radiation releases.	3.8	71		
	2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	3.4	72		
	Subtotal			**2**		
4. Emergency Procedures / Plan	2.4.2	Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	4.5	73		
	2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	4.0	74		
	2.4.45	Ability to prioritize and interpret the significance of each annunciator or alarm.	4.1	75		
	Subtotal			**3**		
Tier 3 Point Total				**10**		

Facility: **Grand Gulf Nuclear Station** Date of Exam: **2/10/2020**

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													4	3	7
	2													1	2	3
	Tier Totals													5	5	10
2. Plant Systems	1													3	2	5
	2													2	1	3
	Tier Totals													5	3	8
3. Generic Knowledge and Abilities Categories					1		2		3		4			1 | 2	3 | 4	7
														2 | 1	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 outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it is replaced by a K/A from another Tier 3 category.)

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points, and the SRO-only exam must total 25 points.
3. Systems/evolutions within each group are identified on the outline. Systems or evolutions that do not apply at the facility should be deleted with justification. Operationally important, site-specific systems/evolutions that are not included on the outline should be added. Refer to Section D.1.b of ES-401 for guidance regarding the elimination of inappropriate K/A statements.
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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

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

ES-401							BWR Examination Outline 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	A 2	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					✓		AA2.02 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Reactor power / pressure / and level	4.3	76
295004 Partial or Total Loss of DC Pwr / 6									
295005 Main Turbine Generator Trip / 3									
295006 SCRAM / 1					✓		AA2.03 Ability to determine and/or interpret the following as they apply to SCRAM : Reactor Water Level	4.2	77
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									
295024 High Drywell Pressure / 5									
295025 High Reactor Pressure / 3					✓		EA2.02 Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor Power	4.2	78
295026 Suppression Pool High Water Temp. / 5									
295027 High Containment Temperature / 5						✓	2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	4.2	79
295028 High Drywell Temperature / 5									
295030 Low Suppression Pool Wtr Lvl / 5									
295031 Reactor Low Water Level / 2					✓		EA2.03 Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : Reactor Pressure	4.2	80
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1									
295038 High Off-site Release Rate / 9						✓	2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	4.7	81
600000 Plant Fire On Site / 8						✓	2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	4.1	82
700000 Generator Voltage and Electric Grid Disturbances / 6									
K/A Category Totals:	**0**	**0**	**0**	**0**	**4**	**3**	**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	K1	K2	K3	A1	A2	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									
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1									
295017 High Off-site Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7					✓		AA2.02 Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION : Drywell/containment temperature	3.4	83
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						✓	2.2.12 Knowledge of surveillance procedures.	4.1	84
295034 Secondary Containment Ventilation High Radiation / 9									
295035 Secondary Containment High Differential Pressure / 5									
295036 Secondary Containment High Sump/Area Water Level / 5									
500000 High CTMT Hydrogen Conc. / 5						✓	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.	4.2	85
K/A Category Point Totals:	**0**	**0**	**0**	**0**	**1**	**2**	**Group Point Total:**		**3**

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

Plant Systems - Tier 2/Group 1 (SRO)

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														
20700 Isol Condenser														
209001 LPCS								✓				A2.06 Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Inadequate system flow	3.2	86
209002 HPCS														
211000 SLC								✓				A2.03 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 or operations: A.C. power failures	3.4	87
212000 RPS								✓				A2.02 Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RPS bus power supply failure	3.9	88
215003 IRM														
215004 Source Range Monitor														
215005 APRM / LPRM														
217000 RCIC														
218000 ADS														
223002 PCIS/Nuclear Steam Supply Shutoff														
239002 SRVs														
259002 Reactor Water Level Control											✓	2.4.50 Ability to verify system alarm setpoints and operate controls identified in | the alarm response manual.	4.0	89
261000 SGTS														
262001 AC Electrical Distribution											✓	2.4.41 Knowledge of the emergency action level thresholds and classifications.	4.6	90
262002 UPS (AC/DC)														
263000 DC Electrical Distribution														
264000 EDGs														
300000 Instrument Air														
400000 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**

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

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	#
201001 CRD Hydraulic								✓				A2.08 Ability to (a) predict the impacts of the following on the CONTROL ROD DRIVE HYDRAULIC SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Inadequate system flow	2.8	91
201002 RMCS														
201003 Control Rod and Drive Mechanism														
201004 RSCS														
201005 RCIS														
201006 RWM														
202001 Recirculation														
202002 Recirculation Flow Control														
204000 RWCU														
214000 RPIS														
215001 Traversing In-Core Probe														
215002 RBM														
216000 Nuclear Boiler Inst.														
219000 RHR/LPCI: Torus/Pool Cooling Mode														
223001 Primary CTMT and Aux.											✓	2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	4.0	92
226001 RHR/LPCI: CTMT Spray Mode														
230000 RHR/LPCI: Torus/Pool Spray Mode														
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment														
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator														
245000 Main Turbine Gen. / Aux.														
256000 Reactor Condensate														
259001 Reactor Feedwater								✓				A2.03 Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:	3.6	93
268000 Radwaste														
271000 Offgas														
272000 Radiation Monitoring														
286000 Fire Protection														
288000 Plant Ventilation														
290001 Secondary CTMT														
290003 Control Room HVAC														
290002 Reactor Vessel Internals														
204000 RWCU														
K/A Category Point Totals:	**0**	**0**	**0**	**1**	**0**	**0**	**0**	**2**	**0**	**0**	**1**	**Group Point Total:**		**3**

Facility: **Grand Gulf Nuclear Station** Date of Exam: **2/10/2020**

Category	K/A #	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct of Operations	2.1.2	Knowledge of operator responsibilities during all modes of plant operation.			4.4	94
	2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management.			4.6	95
	Subtotal					**2**
2. Equipment Control	2.2.7	Knowledge of the process for conducting special or infrequent tests.			3.6	96
	2.2.21	Knowledge of pre- and post-maintenance operability requirements.			4.1	97
	Subtotal					**2**
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits under normal or emergency conditions.			3.7	98
	Subtotal					**1**
4. Emergency Procedures / Plan	2.4.29	Knowledge of the emergency plan.			4.4	99
	2.4.42	Knowledge of emergency response facilities.			3.8	100
	Subtotal					**2**
Tier 3 Point Total						**7**

Tier/Group (Original)	Randomly Selected K/A (New)	Reason for Rejection
RO T1 – G1 295003 AK3.04 Q# 2	295003 AK3.07	Original KA: AK3.04 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Ground isolation At GGNS, if a ground occurred on an A.C. Powered bus the electrical department would perform the check and isolation of the ground. Operations does not perform a ground isolation. Randomly selected New K/A: AK3.05, Reactor SCRAM Page 1 point totals not affected by this change.
RO T1 – G1 295021 AK3.04 Q# 9	295021 AK3.01	Original KA: AK3.04 Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING : Increasing drywell cooling At GGNS, there is no procedural requirement to increase drywell cooling on a loss of Shutdown Cooling. The Off Normal Event Procedure for Inadequate Decay Heat Removal does not mention increasing drywell cooling to mitigate this loss. Randomly selected New K/A: AK3.01, AK3.01 Raising reactor water level Page 1 point totals not affected by this change.
RO T1 – G1 295037 EA2.03 Q# 18	A2 listed but K2 words are used.	This KA was placed in error and replaced, NO swap required, revised ES-401-1, K/A Topic for 295037 to include the following: Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : EA2.03 SBLC tank level Page 1 point totals not affected by this change.
RO T2 – G1 261000 K1.11 Q# 46	261000 K1.08	This KA was placed in error from a different K/A system and replaced. Randomly selected New K/A: K1.08. Knowledge of the physical connections and/or cause effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: Process radiation monitoring system. Page 1 point totals not affected by this change.

RO T2 – G1 510000 K1.02 Q# 53	 209002 K1.02	This KA was placed in error from Rev. 3 NUREG 1123 and replaced. Randomly Selected new K/A 209002: K1.02 Knowledge of the physical connections and/or cause effect relationships between HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) and the following: Suppression Pool: BWR-5,6 Page 1 point totals not affected by this change.
RO T2 – G2 202001 K1.14 Q# 55	 202001 K1.15	Original KA: K1.14 Knowledge of the physical connections and/or cause effect relationships between RECIRCULATION SYSTEM and the following: Rod block monitor At GGNS a Rod Block Monitor system is not used. The RC&IS system provides the Control Rod Block feature. Randomly selected New K/A: K1.15 Raising reactor water level Page 1 point totals not affected by this change.
RO T2 – G2 202002 K6.06 Q# 56	 202002 K6.05	Original KA: K6.06 Knowledge of the effect that a loss or malfunction of the following will have on the RECIRCULATION FLOW CONTROL SYSTEM : Reactor/turbine pressure regulating system At GGNS, the Reactor/turbine pressure regulating system no longer provides any input into the Recirculation Flow Control system anymore. GGNS controls the flow control system in MANUAL valve control, NO auto feedback from the Reactor/turbine pressure regulating system. Randomly selected New K/A: K6.05, Reactor water level Page 1 point totals not affected by this change.
RO T2 – G2 214000 K3.03 Q# 57	 201005 K3.01	Original KA: 214000, K3.03 Knowledge of the effect that a loss or malfunction of the ROD POSITION INFORMATION SYSTEM will have on following: RMCS: Plant-Specific At GGNS, Rod position and information is a part of the RCIS system (K/A 201005). RPIS is not a stand alone system. Randomly selected New K/A: 201005 RCIS K3.01, Knowledge of the effect that a loss or malfunction of the ROD CONTROL AND INFORMATION SYSTEM (RCIS) will have on following: Control rod drive system: BWR-6 Page 1 point totals not affected by this change.

RO T2 – G2 259001 K5.02 Q# 62	 259001 K5.03	Original KA: 259001, K5.02 Knowledge of the operational implications of the following concepts as they apply to REACTOR FEEDWATER SYSTEM : Water hammer At GGNS, Water hammer is a generic concern and not a specific one in the feedwater procedures. Randomly selected New K/A: K5.03, Turbine operation: TDRFP's-Only Page 1 point totals not affected by this change.
RO T2 – G2 272000 A3.05 Q# 63	 272000 A3.06	Original KA: 272000 Radiation Monitoring, A3.05 Ability to monitor automatic operations of the RADIATION MONITORING SYSTEM including: Refuel floor overhead crane operation interrupt At GGNS, there is NO refuel floor overhead crane operation interrupt action with the Radiation Monitoring system. Randomly selected New K/A: A3.06, Ventilation system isolation indications Page 1 point totals not affected by this change.
RO T2 – G2 51001 2.4.49 Q# 65	 219000 2.4.49	This KA was placed in error from Rev. 3 NUREG 1123 and replaced. Randomly Selected new K/A 219000 K/A Topic, 2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls, remains the same. Page 1 point totals not affected by this change.
RO T3 2.1.6 Q# 66	 2.1.8	Original KA: Ability to manage the control room crew during plant transients. The ability to manage as inherent SRO / Supervisor task. Randomly Selected new K/A 2.1.8 K/A Topic, 2.1.8 Ability to coordinate personnel activities outside the control room. Page 1 point totals not affected by this change.

SRO T1–G1 295027 2.2.39 Q# 79	 295027 2.2.25	Original KA: 2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems. This K/A does not meet the knowledge requirements for a SRO, <1 hour Tech Specs action statements are RO knowledge level. Randomly selected New K/A: 2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. Page 1 point totals not affected by this change.
SRO T1–G1 510000 2.4.41 Q# 90	 262001 2.4.41	This KA was placed in error from Rev. 3 NUREG 1123 and replaced. Randomly Selected new K/A 262001. 2.4.41 Knowledge of the emergency action level thresholds and classifications, remained the same. Page 1 point totals not affected by this change.

Facility: Grand Gulf Nuclear Station Date of Examination: 2/3/2020

Examination Level: RO ☒ SRO ☐ Operating Test Number: GGNS 2/2020

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	**R, N**	**Determine Core Flow**, given plant parameters, system status or events determine actual Reactor Core Flow. GJPM-OPS-2/2020-AR1 2.1.7 (4.4)
Conduct of Operations	**R, D**	**Reactor Water Level Determination,** The operator will evaluate plant conditions at the Remote Shutdown Panel to determine Narrow Range reactor water level and the status of injection sources. GJPM-OPS-2/2020-AR2 2.1.25 (3.9)
Radiation Control	**R, D**	**Emergency Exposure Limits,** The operator will evaluate a condition involving abnormally high radiological conditions and determine actions required to administratively control the dose received by determining who authorizes dose extensions in various situations. GJPM-OPS-2/2020-AR3 2.3.4 (3.2)
Emergency Plan	**R, N**	**Loss of Shutdown Cooling, Time to 200°F,** given plant status and parameters determine the time to 200 on a loss of Shutdown Cooling. GJPM-OPS-2/2020-AR4 2.4.11 (3.4/3.6)

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

* Type Codes and Criteria:

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

Facility: Grand Gulf Nuclear Station Date of Examination: 2/3/2020

Examination Level: RO ☐ SRO ☒ Operating Test Number: GGNS 2/2020

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	**R, D**	**Determine Reportability**, This task is to use corporate procedures and given plant conditions to determine the reportability and complete applicable notification form GJPM-OPS-2/2020-AS1 2.1.20 (4.6)
Conduct of Operations	**R, D**	**Determine Penetration Isolation Requirements,** given failure of penetration isolation valve failure, determine Tech Spec action and requirements for isolating. GJPM-OPS-2/2020-AS2 2.1.7 (4.7)
Equipment Control	**R, D**	**Tagout Removal Approval,** given a protective tagging removal, verify restored positions and information. GJPM-OPS-2/2020-AS3 2.2.15 (4.3) – 2.2.15 (4.3) – 2.2.41 (3.9)
Radiation Control	**R, D**	**Authorize Emergency Exposure,** given plant status and events in progress, determine appropriate radiation exposure limits. GJPM-OPS-2/2020-AS4 2.3.4 (3.7)
Emergency Plan	**R, N**	**Emergency Classification,** given plant status and parameters determine the correct emergency classification. GJPM-OPS-2/2020-AS5 2.4.41 (4.6)

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

* Type Codes and Criteria:

- (C)ontrol room, (S)imulator, or Class(R)oom
- (D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs and RO retakes) **(4)**
- (N)ew or (M)odified from bank (≥ 1) **(1)**
- (P)revious 2 exams (≤ 1, randomly selected) **(0)**

Facility: **Grand Gulf Nuclear Station** Date of Examination: **2/03/2020**

Exam Level: **RO** ☒ SRO-I ☐ SRO-U ☐ Operating Test No.: **GGNS 2-2020**

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

System / JPM Title	Type Code*	Safety Function
S1 – Transfer Recirculation Pumps to Slow Speed (GJPM-OPS-2-2020S1) 202001 A2.04 & A4.01 (3.7)	**A**-D-S	1
S2 – Retest MSIV Slow Closure (GJPM-OPS-2-2020S2) 239001 A2.11 (4.1)	**A**-D-P-S	3
S3 – Performing HPCS Quarterly Functional Test (GJPM-OPS-2-2020S3) 209002: A1.01 (3.6); A4.01 (3.7)	**A**-D-S-EN	4
S4 – Secure Containment Spray and Align for RPV Injection (GJPM-OPS-2-2020S4) 226001 A2.20 (3.7) & A4.07 (3.5)	L-M-S-EN	5
S5 – Rotate CCW Pumps (GJPM-OPS-2-2020S5) 400000 A2.01 (3.3) & A4.01 (3.1)	**A**-D-S	8
S6 – Place Standby Gas Treatment System in STANDBY Mode (GJPM-OPS-2-2020S6) 261000 A2.13 (3.4) & A4.03 (3.0)	L-N-S-EN	9
C1 – Defeat Feed Pump Level 9 Trips (GJPM-OPS-2-2020CR1) 259001 A3.10 (3.4)	D-C-L	2
S7 – Transfer RPS B to Normal Power Source and RPS A to Alternate Power Source **(RO ONLY)** (GJPM-OPS-2-2020S7) 212000 A2.19 (3.8) & A4.14 (3.8)	D-S	7
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
P1 – RPS Motor Generator Startup (GJPM-OPS-2-2020P1) 212000 A2.01 (3.7) & A1.01 (2.8)	D	7
P2 – Align Fire Water to RHR 'C' per EP Attachment 26 (GJPM-OPS-2-2020P2) 286000 A1.05 (3.2)	D-E-L-R	8
P3 – HPCS Diesel Generator Emergency Shutdown (GJPM-OPS-2-2020P3) 264000 A4.04 (3.7)	**A**-E-N	6

* 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	**A**	**4-6** / 4-6 / 2-3	**(5)**
(**C**)ontrol room	**C**	-----	**(1)**
(**D**)irect from bank	**D**	**≤ 9** / ≤ 8 / ≤ 4	**(8)**
(**E**)mergency or abnormal in-plant	**E**	**≥ 1** / ≥ 1 / ≥ 1	**(2)**
(**EN**)gineered safety feature	**EN**	**≥ 1** / ≥ 1 / ≥ 1 (control room sys)	**(3)**
(**L**)ow-Power / Shutdown	**L**	**≥ 1** / ≥ 1 / ≥ 1	**(3)**
(**N**)ew or (**M**)odified from bank including 1(**A**)	**N-M**	**≥ 2** / ≥ 2 / ≥ 1	**(3)**
(**P**)revious 2 exams	**P**	**≤ 3** / ≤ 3 / ≤ 2 (randomly selected)	**(1)**
(**R**)CA	**R**	**≥ 1** / ≥ 1 / ≥ 1	**(1)**
(**S**)imulator	**S**		**(7)**

JPM Description:

S1 – Transfer Recirculation Pumps to Slow Speed
(GJPM-OPS-2-2020S1) 202001 A2.04 & A4.01 (3.7)

- This task is to transfer Reactor Recirculation Pumps to slow speed per IOI - 2, 03-1-01-2, Power Operations Attachment IV section 8.8.6 and 8.8.7. Upon transfer to slow speed and reopening of both Recirculation Flow Control Valves to 50%, both Reactor Recirculation Pumps will trip to OFF. With the Reactor Mode Switch in RUN at power and no Reactor Recirculation Pumps operating, entry into the Reduction in Recirculation Flow ONEP 05-1-02-III-3, Immediate Operator Actions step 2.1.2 will be required to insert a Manual Reactor Scram.

- Placing the Reactor Mode Switch to Shutdown will NOT insert Control Rods requiring alternate actions to insert the Control Rods by either arming and depressing the Manual Scram pushbuttons OR Initiating ATWS ARI/RPT which will depressurize the Scram Air Header.

S2 – Retest MSIV Slow Closure
(GJPM-OPS-2-2020S2) 239001 A2.11 (4.1)

- This task is to perform a slow closure on MSIV B21-F028A.

- Following the closure of the MSIV, recognize High - High temperature in Main Steam Tunnel without Group 1 MSIV isolation. Applicant should manually close all remaining MSIVs (7) using handswitches on P601 panel to isolate the steam leak.

- At initial power level Main Steam Line Drains should be closed with the exception of B21-F019, INBD MSL DR OTBD DR VLV which has another valve in the line already closed B21-F016, INBD MSL DR INBD DR VLV.

S3 – Performing HPCS Quarterly Functional Test
(GJPM-OPS-2-2020S3) 209002: A1.01 (3.6); A4.01 (3.7)

- This task requires the ability to manually start the only ECCS-qualified high pressure injection system.
- This task demonstrates the ability to operate HPCS in the "test return" mode, which puts HPCS flow in a loop from and to the Suppression Pool, one of its two suction sources. HPCS is operated in this mode for surveillance and post-maintenance testing.
- As HPCS is placed in the test return mode the system will experience a failure to initiate and a failure to inject once manually initiated. This will require to arm and depress the initiation pushbutton and then manually open the E22-F004.

S4 – Secure Containment Spray and Align for RPV Injection
(GJPM-OPS-2-2020S4) 226001 A2.20 (3.7) & A4.07 (3.5)

- This task is to secure RHR systems from Containment Spray and align them for injection into the RPV during a LOCA. During the performance one RHR system will not provide sufficient flow to raise RPV water level and the second RHR system will have a failure of E12-F042 LPCI injection valve to open requiring the use of an alternate injection path through E12-F053. Realignment of RHR from Containment Spray to LPCI mode is directed from the Emergency Procedures when there is not Adequate Core Cooling. Use of Shutdown Cooling lines for injection to the RPV from RHR is allowed per the Emergency Procedures and attachments are provided to facilitate this evolution.

S5 – Rotate CCW Pumps
(GJPM-OPS-2-2020S5) 400000 A2.01 (3.3) & A4.01 (3.1)

- This task is to rotate CCW Pumps per SOI. During the evolution, a trip will occur on one of the operating CCW pumps requiring the restart of the non-operating CCW pump per the Loss of CCW ONEP.

S6 – Place Standby Gas Treatment System in STANDBY Mode
(GJPM-OPS-2-2020S6) 261000 A2.13 (3.4) & A4.03 (3.0)

- This task is to place one Standby Gas Treatment System in STANDBY Mode after an automatic initiation signal on Radiation that is still present.

C1 – Defeat Feed Pump Level 9 Trips
(GJPM-OPS-2-2020CR1) 259001 A3.10 3.4/3.4

- This task defeats the High Reactor Water Level trip of the Reactor Feed Pumps, which under certain conditions is directed by the EOPs to maintain adequate core cooling.

S7 – Transfer RPS B to Normal Power Source and RPS A to Alternate Power Source **(RO ONLY)**
(GJPM-OPS-2-2020S7) 212000 A2.19 (3.8) & A4.14 (3.8)

- This task is to align RPS B power to be supplied from its Normal source, the Motor Generator Set, and to align RPS A power to be supplied from its Alternate source, 480V ESF breaker 52-154204.

P1 – RPS Motor Generator Startup
(GJPM-OPS-2-2020P1) 212000 A2.01 (3.7) & A1.01 (2.8)

- This task is to perform a startup of the RPS Motor Generator and align the RPS Bus to the Normal Supply per the SOI.

P2 – Align Fire Water to RHR ‘C’ per EP Attachment 26
(GJPM-OPS-2-2020P2) 286000 A1.05 (3.2)

- This task simulates routing and connecting fire hoses from hose stations to test connections on ECCS injection piping in the Auxiliary Building.

P3 – HPCS Diesel Generator Emergency Shutdown
(GJPM-OPS-2-2020P3) 264000 A4.04 (3.7)

- This task simulates an auto start of HPCS Diesel Generator with a subsequent oil system failure requiring the operator to EMERGENCY STOP (trip) the HPCS D/G. The Control Room switch and local Emergency Stop switch will not work, the operator must use other means to stop the EDG. Per P81 SOI the lay shaft handle on each diesel engine must be manipulated to stop the EDG.

Facility: **Grand Gulf Nuclear Station**	Date of Examination: **2/03/2020**
Exam Level: RO ☐ **SRO-I ☒** SRO-U ☐	Operating Test No.: **GGNS 2-2020**

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

System / JPM Title	Type Code*	Safety Function
S1 – Transfer Recirculation Pumps to Slow Speed (GJPM-OPS-2-2020S1) 202001 A2.04 & A4.01 (3.7)	**A**-D-S	1
S2 – Retest MSIV Slow Closure (GJPM-OPS-2-2020S2) 239001 A2.11 (4.1)	**A**-D-P-S	3
S3 – Performing HPCS Quarterly Functional Test (GJPM-OPS-2-2020S3) 209002: A1.01 (3.6); A4.01 (3.7)	**A**-D-S-EN	4
S4 – Secure Containment Spray and Align for RPV Injection (GJPM-OPS-2-2019S4) 226001 A2.20 (3.7) & A4.07 (3.5)	L-M-S-EN	5
S5 – Rotate CCW Pumps (GJPM-OPS-2-2020S5) 400000 A2.01 (3.3) & A4.01 (3.1)	**A**-D-S	8
S6 – Place Standby Gas Treatment System in STANDBY Mode (GJPM-OPS-2-2020S6) 261000 A2.13 (3.4) & A4.03 (3.0)	L-N-S-EN	9
C1 – Defeat Feed Pump Level 9 Trips (GJPM-OPS-2-2020CR1) 259001 A3.10 (3.4)	D-C-L	2
In-Plant Systems* (3 for RO); **(3 for SRO-I);** (3 or 2 for SRO-U)		
P1 – RPS Motor Generator Startup (GJPM-OPS-2-2020P1) 212000 A2.01 (3.7) & A1.01 (2.8)	D	7
P2 – Align Fire Water to RHR 'C' per EP Attachment 26 (GJPM-OPS-2-2020P2) 286000 A1.05 (3.2)	D-E-L-R	8
P3 – HPCS Diesel Generator Emergency Shutdown (GJPM-OPS-2-2020P3) 264000 A4.04 (3.7)	**A**-E-N	6

* 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	**A**	4-6 / **4-6** / 2-3	**(5)**
(**C**)ontrol room	**C**	-----	**(1)**
(**D**)irect from bank	**D**	≤ 9 / **≤ 8** / ≤ 4	**(7)**
(**E**)mergency or abnormal in-plant	**E**	≥ 1 / **≥ 1** / ≥ 1	**(2)**
(**EN**)gineered safety feature	**EN**	≥ 1 / **≥ 1** / ≥ 1 (control room sys)	**(3)**
(**L**)ow-Power / Shutdown	**L**	≥ 1 / **≥ 1** / ≥ 1	**(3)**
(**N**)ew or (**M**)odified from bank including 1(**A**)	**N-M**	≥ 2 / **≥ 2** / ≥ 1	**(3)**
(**P**)revious 2 exams	**P**	≤ 3 / **≤ 3** / ≤ 2 (randomly selected)	**(1)**
(**R**)CA	**R**	≥ 1 / **≥ 1** / ≥ 1	**(1)**
(**S**)imulator	**S**		**(6)**

JPM Description:

S1 – Transfer Recirculation Pumps to Slow Speed
(GJPM-OPS-2-2020S1) 202001 A2.04 & A4.01 (3.7)

- This task is to transfer Reactor Recirculation Pumps to slow speed per IOI - 2, 03-1-01-2, Power Operations Attachment IV section 8.8.6 and 8.8.7. Upon transfer to slow speed and reopening of both Recirculation Flow Control Valves to 50%, both Reactor Recirculation Pumps will trip to OFF. With the Reactor Mode Switch in RUN at power and no Reactor Recirculation Pumps operating, entry into the Reduction in Recirculation Flow ONEP 05-1-02-III-3, Immediate Operator Actions step 2.1.2 will be required to insert a Manual Reactor Scram.

- Placing the Reactor Mode Switch to Shutdown will NOT insert Control Rods requiring alternate actions to insert the Control Rods by either arming and depressing the Manual Scram pushbuttons OR Initiating ATWS ARI/RPT which will depressurize the Scram Air Header.

S2 – Retest MSIV Slow Closure
(GJPM-OPS-2-2020S2) 239001 A2.11 (4.1)

- This task is to perform a slow closure on MSIV B21-F028A.

- Following the closure of the MSIV, recognize High - High temperature in Main Steam Tunnel without Group 1 MSIV isolation. Applicant should manually close all remaining MSIVs (7) using handswitches on P601 panel to isolate the steam leak.

- At initial power level Main Steam Line Drains should be closed with the exception of B21-F019, INBD MSL DR OTBD DR VLV which has another valve in the line already closed B21-F016, INBD MSL DR INBD DR VLV.

S3 – Performing HPCS Quarterly Functional Test
(GJPM-OPS-2-2020S3) 209002: A1.01 (3.6); A4.01 (3.7)

- This task requires the ability to manually start the only ECCS-qualified high pressure injection system.
- This task demonstrates the ability to operate HPCS in the "test return" mode, which puts HPCS flow in a loop from and to the Suppression Pool, one of its two suction sources. HPCS is operated in this mode for surveillance and post-maintenance testing.
- As HPCS is placed in the test return mode the system will experience a failure to initiate and a failure to inject once manually initiated. This will require to arm and depress the initiation pushbutton and then manually open the E22-F004.

S4 – Secure Containment Spray and Align for RPV Injection
(GJPM-OPS-2-2020S4) 226001 A2.20 (3.7) & A4.07 (3.5)

- This task is to secure RHR systems from Containment Spray and align them for injection into the RPV during a LOCA. During the performance one RHR system will not provide sufficient flow to raise RPV water level and the second RHR system will have a failure of E12-F042 LPCI injection valve to open requiring the use of an alternate injection path through E12-F053. Realignment of RHR from Containment Spray to LPCI mode is directed from the Emergency Procedures when there is not Adequate Core Cooling. Use of Shutdown Cooling lines for injection to the RPV from RHR is allowed per the Emergency Procedures and attachments are provided to facilitate this evolution.

S5 – Rotate CCW Pumps
(GJPM-OPS-2-2020S5) 400000 A2.01 (3.3) & A4.01 (3.1)

- This task is to rotate CCW Pumps per SOI. During the evolution, a trip will occur on one of the operating CCW pumps requiring the restart of the non-operating CCW pump per the Loss of CCW ONEP.

S6 – Place Standby Gas Treatment System in STANDBY Mode
(GJPM-OPS-2-2020S6) 261000 A2.13 (3.4) & A4.03 (3.0)

- This task is to place one Standby Gas Treatment System in STANDBY Mode after an automatic initiation signal on Radiation that is still present.

C1 – Defeat Feed Pump Level 9 Trips
(GJPM-OPS-2-2020CR1) 259001 A3.10 3.4/3.4

- This task defeats the High Reactor Water Level trip of the Reactor Feed Pumps, which under certain conditions is directed by the EOPs to maintain adequate core cooling.

P1 – RPS Motor Generator Startup
(GJPM-OPS-2-2020P1) 212000 A2.01 (3.7) & A1.01 (2.8)

- This task is to perform a startup of the RPS Motor Generator and align the RPS Bus to the Normal Supply per the SOI.

P2 – Align Fire Water to RHR ‘C’ per EP Attachment 26
(GJPM-OPS-2-2020P2) 286000 A1.05 (3.2)

- This task simulates routing and connecting fire hoses from hose stations to test connections on ECCS injection piping in the Auxiliary Building.

P3 – HPCS Diesel Generator Emergency Shutdown
(GJPM-OPS-2-2020P3) 264000 A4.04 (3.7)

- This task simulates an auto start of HPCS Diesel Generator with a subsequent oil system failure requiring the operator to EMERGENCY STOP (trip) the HPCS D/G. The Control Room switch and local Emergency Stop switch will not work, the operator must use other means to stop the EDG. Per P81 SOI the lay shaft handle on each diesel engine must be manipulated to stop the EDG.

Facility: **GRAND GULF NUCLEAR STATION** Date of Exam: **2 / 3 / 2020** Operating Test No.:**GGNS 2-2020**

APPLICANT	EVENT TYPE	Scenarios 1 CREW POSITION SRO	1 ATC	1 BOP	2 CREW POSITION SRO	2 ATC	2 BOP	3 CREW POSITION SRO	3 ATC	3 BOP	4 CREW POSITION SRO	4 ATC	4 BOP	TOTAL	MINIMUM(*) R	I	U
RO ☐ SRO-I ■ SRO-U ☐	RX														1	1	0
	NOR	1			1									2	1	1	1
	I/C	5			6			5			4			20	4	4	2
	MAJ	2			2			2			3			9	2	2	1
	TS	2			2			2			2			8	0	2	2
RO ■ SRO-I ☐ SRO-U ☐	RX														1	1	0
	NOR						1							1	1	1	1
	I/C		3				3		3				2	11	4	4	2
	MAJ		2				2		2				3	9	2	2	1
	TS														0	2	2
RO ■ SRO-I ☐ SRO-U ☐	RX														1	1	0
	NOR			1										1	1	1	1
	I/C			3		3				3		3		12	4	4	2
	MAJ			2		2				2		3		9	2	2	1
	TS																
	RX																
	NOR																
	I/C																
	MAJ																
	TS																

Instructions:

1. Check the applicant level and enter the operating test number and Form ES-D-1 event numbers for each event type; TS are not applicable for RO applicants. ROs must serve in both the at-the-controls (ATC) and balance-of-plant (BOP) positions. Instant SROs (SRO-I) must serve in both the SRO and the ATC positions, including at least two instrument or component (I/C) malfunctions and one major transient, in the ATC position. If an SRO-I *additionally* serves in the BOP position, one I/C malfunction can be credited toward the two I/C malfunctions required for the ATC position.

2. Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.5.d) but must be significant per Section C.2.a of Appendix D. (*) Reactivity and normal evolutions may be replaced with additional I/C malfunctions on a one-for-one basis.

3. Whenever practical, both instrument and component malfunctions should be included; only those that require verifiable actions that provide insight to the applicant's competence count toward the minimum requirements specified for the applicant's license level in the right-hand columns.

4. For new reactor facility licensees that use the ATC operator primarily for monitoring plant parameters, the chief examiner may place SRO-I applicants in either the ATC or BOP position to best evaluate the SRO-I in manipulating plant controls.

GGNS 2-2020 NRC Scenario 2

Facility: Grand Gulf Nuclear Station Scenario No.: 2 Op-Test No.: GGNS 2-2020

Examiners: ______________ Operators: ______________

Initial Conditions: 100% power.

Inoperable equipment: None

Turnover: Div 1 work week. Place SSW A in recirculation mode for chemistry sample

Event No.	Malf. No.	Event Type †	Event Description
1	p41f005a_i	N (BOP, CRS) C (BOP, CRS) TS (CRS)	Place SSW A in Recirc for chemistry sample. P41-F005A trip on stroke. LCO 3.7.1, Cond D
2	ftb33n014b_e	C (ATC, CRS)	Recirc suction flow transmitter loses power. PLCO 3.3.1.1 Cond. A
3	p43152a	C (BOP, CRS) A (Crew)	TBCW pump trip and standby pump fails to auto start
4	z025025_60_37 z025025_56_37 z025025_52_37	C (ATC, CRS) A (Crew) TS (CRS)	3 control rods scram in due to air leak LCO 3.1.3, Cond C
5	tc093	M (CREW)	Spurious Main Turbine Trip
6	c11164	M (CREW)	ATWS > 5% power (CT-1) terminates injection to lower Rx level (CT-2) initiates standby liquid control (CT-3) inserts control rods
7	RF Att. 11	C (ATC, CRS)	Defeat RWCU isolation on SLC initiation
8	r21180	C (BOP, CRS)	ESF 21 lockout

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

GGNS 2-2020 NRC Scenario 2

Quantitative Attributes Table			
Attribute	**E3-301-4 Target**	**Actual**	**Description**
Malfunctions after EOP entry	1-2	2	• RWCU fails to isolate on SLC initiation • ESF 21 lockout
Abnormal Events	2-4	4	• SSW A return valve P41F005A trip on stroke • TBCW pump trip and standby pump fails to auto start • B33 Flow Xmitter failure • 3 control rods scram in
Major Transients	1-2	2	• Spurious main turbine trip • ATWS
EOP entries requiring substantive action	1-2	1	• ATWS, EP-2A
Entry into a contingency EOP with substantive actions	≥1	1	• ATWS, EP-2A
Preidentified critical tasks	2-3	3	• (CT-1) When control rods fail to scram, crew inserts all control rods to position 02 or beyond before exiting EP-2A • (CT-2) During failure to scram conditions with power > 5% RTP, terminates feedwater injection within 90 seconds of ARI/RPT initiation and prevents injection from all other sources (except boron, CRD, and RCIC) as necessary to lower RPV level to below -70” wide range prior to exiting EP-2A. • (CT-3) With Reactor Power above 5%, inject Standby Liquid Control system within 5 minutes AND/OR prior to suppression pool temperature reaching 110°F, whichever is later.

GGNS 2-2020 NRC Scenario 2

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

- Respond to a trip of P41-F005A, SSW A return valve when placing SSW A in recirc.
- Respond to a failure of a Rx Recirc suction flow transmitter failure that results in a rod block
- Respond to a trip of a TBCW pump and the standby pump fails to auto start
- Respond to 3 control rods scramming in due to an air leak
- Respond to a spurious turbine trip
- Respond to an ATWS that generates > 5% Rx power
- Respond to a failure of RWCU to isolate on Standby Liquid Control initiation
- Respond to an ESF 21 lockout

Initial Conditions: Plant is operating at 100% power.

Inoperable Equipment: None

Turnover:

- This is a Division 1 work week
- Place SSW A in recirculation mode for chemistry sample

Scenario Notes:

- This scenario is a NEW Scenario.
- Validation Time: 60 minutes

GGNS 2-2020 NRC Scenario 2

SCENARIO ACTIVITIES:

Initial Conditions

Initial Conditions: 100% Power.

Inoperable Equipment: None

Event 1 – P41-F005A trip on stroke (Initial Setup - Automatic)

The BOP operator will place SSW A in recirculation mode for a chemistry sample per SOI 04-1-01-P41-1, section 5.10. The pump discharge relief valve will be open requiring the operator to place flow through the RHR A heat exchanger. When the F005A handswitch is taken to open, its breaker will trip. This will require placing the Div 1 DG in maintenance and racking out the Div 1 DW purge compressor breaker. LCO 3.7.1, Cond D.

Event 2 – Recirc flow transmitter failure (Triggered by Lead Examiner)

A ‘B’ Rx recirculation loop suction transmitter will lose power and input ‘0’ flow into ‘B’ APRM/OPRM trip logic. When investigated, flow input will be half of the other APRM/OPRMs since a similar suction flow transmitter on the A Recirc loop continues to provide input. Per ARI instruction, the APRM should be declared inop, bypassed, and potential LCO 3.3.1.1 identified.

Event 3 – TBCW pump trip with standby pump auto start failure (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. The CRS should enter 05-1-02-V-2, Loss of Turbine Building Cooling Water ONEP and ensure parameters have stabilized.

Event 4 – Three control rods scram in (Triggered by Lead Examiner)

3 control rods will scram in due to an air leak. This will require entry into 05-1-02-IV-1, Control Rod/Drive Malfunctions ONEP and require reducing Recirc flow to 70 mlbm/hr. Lowering Recirc flow will require entry into 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP to ensure Rx and Recirc system parameters are as expected. If an operator is dispatched to investigate, he will simulate isolating the air leak which will allow 2 of the 3 control rods to settle. LCO 3.1.3, Cond C should be identified for the inop control rod(s).

There is a potential this will be misdiagnosed as 3 control rods drifting in, which requires a manual scram.

GGNS 2-2020 NRC Scenario 2

<u>Event 5 – Spurious Main Turbine Trip</u> (Triggered by Lead Examiner)

A spurious main turbine trip will occur. The automatic actions of the turbine trip ONEP should be verified.

<u>Event 6 - Hydraulic Block ATWS > 5% RTP</u> (Initial Setup - Automatic)

When the reactor scrams due to the turbine trip, an ATWS occurs due to a hydraulic block of both scram discharge volumes. EP-2A is entered via EP-2. Reactor power will be above 5% RTP. Crew will install the necessary attachments to bypass RPS and RC&IS interlocks and insert controls rods via manual scrams and RC&IS **(CT-3)**. Terminate and Prevent is required because reactor power is above 5% RTP. RPV level is intentionally lowered below -70 inches wide range in order to lower core inlet subcooling which prevents/suppresses instabilities and lowers reactor power **(CT-1)**. Bypass valves will control reactor pressure. Turbine pressure setpoint should be lowered to 900 psig to lower RPV pressure below the reset value for low-low set valves.

<u>Event 7 – RWCU fails to auto isolate on SLC injection</u> (Initial Setup - Automatic)

RWCU fails to auto isolate on level 2 (-41.6”) or SLC initiation. When the immediate action to inject SLC is performed **(CT-2)**, (a minimum) 1 RWCU MOV will be required to be closed to prevent RWCU from removing SLC from the RPV. With SLC injection, Rx power will lower. Based in the ATWS power level, sufficient SLC content is being injected and combined with rod insertion, the Rx will be shutdown.

<u>Event 8 – ESF 21 lockout</u> (Initial Setup - Automatic)

ESF 21 will lockout. The Div 2 DG will automatically tie to 16AB and Div 3 DG will automatically tie to 17AC. This will require a re-terminate/prevent of Div 2 ECCS and restoration of CRD pump B.

<u>Termination:</u>

A. Once control rods are being inserted and 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.

GGNS 2-2020 NRC Scenario 2

Critical Task	(CT-1) During failure to scram conditions with power > 5%, • terminate feedwater injection to lower RPV level to below -70" wide range within 90 seconds following ATWS ARI/RPT initiation. This is a time critical action. • terminate and prevent all other injection sources (except boron, CRD, and RCIC) as necessary to lower RPV level to below -70" wide range prior to exiting EP-2A.
Event	6
Safety Significance	Regarding lowering level below -70" wide range, to prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities. RPV water level is lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, the initiation and growth of oscillations is principally dependent upon the subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude. 24" below the lowest nozzle in the feedwater sparger has been selected as the upper bound of the RPV water level control band. This water level is sufficiently low that steam heating of the injected water will be at least 65% to 75% effective (i.e., the temperature of the injected water will be increased to 65% to 75% of its equilibrium value in the steam environment). This water level is sufficiently high that most plants without the capability to readily defeat the low RPV water level MSIV isolation should be able to control RPV water level with feedwater pumps to preclude the isolation.
Cueing	A scram is initiated (either automatically or manually) and numerous control rods indicate beyond position 02 and reactor power is > 5% on panel P680 indications and SPDS and RPV level is > -70" wide range on SPDS and PDS.

Critical Task	(CT-2) With Reactor Power above 5% inject Standby Liquid Control system within 5 minutes OR prior to suppression pool temperature reaching 110°F.
Event	6
Safety Significance	GGNS-NE-16-00004, Time Critical Operator Actions for Grand Gulf, Attachment 1 02-S-01-40, EP Technical Bases, Attachment IX Steps Q2 – Q4 If reactor power remains above the APRM downscale setpoint (5%) following multiple attempts to scram the reactor (EP-2 Step 1, EP-2A Step 1), boron injection is initiated immediately to preclude power oscillations, avoid challenges to primary containment temperature and pressure limits, and ensure that the plant remains in a controlled state. SLC is the normal method of injecting boron into the RPV. Both pumps are started to increase the injection rate and shorten the time required to complete boron injection.
Cueing	Operator verifies SLC is injecting IAW 04-1-01-C41-1, Attachment VI, Verification of Standby Liquid Control Injection.

GGNS 2-2020 NRC Scenario 2

Critical Task	(CT-3) When control rods fail to scram, crew inserts all control rods to position 02 or beyond before exiting EP-2A
EVENT	6
Safety Significance	Failure to effect shutdown of the reactor when a RPS setpoint has been exceeded would unnecessarily extend the level of degradation of the safety of the plant. This could further degrade into damage to the principle fission product barriers if left unmitigated. The crew is authorized by Conduct of Operations to take mitigating actions when automatic safety systems fail to perform their intended function. Action to shut down the reactor is required when RPS and control rod drive systems fail IAW EP-2A.
Cueing	• A scram is initiated (either automatically or manually) and numerous control rods indicate beyond position 02.

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

GGNS 2-2020 NRC Scenario 4

Facility: Grand Gulf Nuclear Station Scenario No.: 4 Op-Test No.: GGNS 2-2020

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Initial Conditions: 79% power.

Inoperable equipment: Div 1 diesel generator, E30-F001A

Turnover: Sequence exchange completed last shift. Reactor Engineering is verifying rod pattern and thermal limits prior to raising core flow.

Event No.	Malf. No.	Event Type †	Event Description
1	c41f004b_a	TS (CRS)	SLC B squib valve failure LCO 3.1.7, Cond D With Div 1 DG inop, LCO 3.1.7, Cond E in 4 hours
2	n34098	C (ATC, CRS) A (ALL)	Turbine lube oil temp controller failure
3	e51188	I (BOP, CRS) A (ALL) TS (CRS)	Spurious RCIC initiation (TS) LCO 3.5.3, Cond A LCO 3.3.5.2, Cond B
4	r21139d	C (BOP, ATC, CRS) A (ALL)	28AG lockout
5	fw203 rr190b c71076	M (CREW)	Recirc pumps downshift / LFMG trip / THI
6	rr063a	M (CREW)	LOCA
7	fw171a b21f065a_i	M (CREW)	Feedwater line break in DW, F065A isolation valve trips on stroke.
8	ct218e ct219b O/Rs e30f002b stem/disc separation	C (ATC, CRS)	HPCS supp pool leak / Door failure / SPMU failure / emergency depressurization. (CT-1) (CT-2)
†	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec		

GGNS 2-2020 NRC Scenario 4

Quantitative Attributes Table			
Attribute	**E3-301-4 Target**	**Actual**	**Description**
Malfunctions after EOP entry	1-2	2	• HPCS suppression pool leak / door failure • E30-F002B stem/disc separation
Abnormal Events	2-4	3	• Turbine lube oil temp controller failure • Spurious RCIC initiation • 28AG lockout
Major Transients	1-2	3	• Recirc pumps downshift / 1 LFMG trip / THI • LOCA • Feedwater line break in DW / inability to isolate
EOP entries requiring substantive action	1-2	2	• LOCA, EP-2 • HPCS Suppression pool leak / door failure, EP-3
Entry into a contingency EOP with substantive actions	≥1	2	• Alternate level control, EP-2 • Emergency Depressurization, EP-2
Preidentified critical tasks	2-3	2	(CT-1) Emergency Depressurize the RPV prior to Suppression Pool level reaching 14.5 ft. (CT-2) After Emergency Depressurization, restore and maintain RPV level above -191” using available injection systems prior to exiting EP-2.

GGNS 2-2020 NRC Scenario 4

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

- Respond to a SLC squib valve loss of power
- Respond to a turbine lube oil temperature controller failure
- Respond to a spurious RCIC initiation
- Respond to a 28AG lockout
- Respond to a double recirc pump downshift / 1 LFMG failure / THI / failure to auto scram
- Respond to a LOCA
- Respond to a feedwater line break in the DW that cannot be isolated
- Respond to a suppression pool leak into HPCS with door failure and SPMU failure

Initial Conditions: Plant is operating at 100% power.

Inoperable Equipment:

- Div 1 diesel generator
- E30-F001A

Turnover:

- This is a Division 1 work week

Scenario Notes:

- This scenario is a NEW Scenario.
- Validation Time: 60 minutes

GGNS 2-2020 NRC Scenario 4

SCENARIO ACTIVITIES:

Initial Conditions

Initial Conditions: 100% Power.

Inoperable Equipment:

- Div 1 diesel generator
- E30-F001A

Event 1 – SLC B squib valve loss of power (Triggered by Lead Examiner)

With Div 1 DG inop and with B SLC inop, LCO 3.1.7, Cond D. With Div 1 DG inop, LCO 3.1.7, Cond E in 4 hours

Event 2 – Turbine lube oil temperature controller failure failure (Triggered by Lead Examiner)

The controller should be placed in manual IAW ARI P680-10A-D3 and restore lube oil temperature to 113°F.

Event 3 – Spurious RCIC initiation (Triggered by Lead Examiner)

RCIC will initiate due to a spurious level 2 initiation signal. The BOP operator will secure RCIC either by tripping RCIC or securing per the hardcard by closing the trip/throttle valve, placing its controller in manual, and reducing its controller output to minimum. With either case, RCIC should be declared inop since it cannot automatically start / inject.

Event 4 – 28AG lockout (Triggered by Lead Examiner)

(reference CR GG-2019-3822)

Based on alarms received, the crew should recognize a loss of multiple PSW pumps due to 28AG lockout. The crew should enter 05-1-02-I-4, Loss of AC Power ONEP and 05-1-02-V-11, Loss of Plant Service Water ONEP. The bus is locked out and the crew should identify a computer point from the Loss of AC power ONEP and determine the bus cannot be reenergized. With core flow already at 70Mlbm/hr, control rods should be inserted to achieve < 50% Rx power. The BOP operator should start the (one) available PSW pump and optimize flow from all operating pumps.

GGNS 2-2020 NRC Scenario 4

Event 5 – Recirc pumps downshift / LFMG failure / THI / auto scram failure

(Initial Setup - Automatic)

When the 4th control rod is inserted, both of the recirc pumps will downshift due to faulty feedwater flow input to the cavitation interlock circuitry. The B LFMG will trip 40 seconds later. 60 seconds later, THI will start ramping in and will result in receipt of P680 alarms indicating the Rx should have automatically scrammed (5A-A11, 5A-B11, 7A-A11, 7A-B11). The ATC operator should insert a manual scram (EP-2 entry).

Event 6 - LOCA (Initial Setup - Automatic)

After the reactor is scrammed, a recirc suction leak will occur resulting in high drywell pressure ECCS initiations and isolations.

Event 7 – Feedwater break in drywell (Initial Setup - Automatic)

A feedwater line A break will occur in the drywell. An attempt should be made to isolate the leak by closing FW INL SHUTOFF VLV, B21-F065A, but the breaker will trip on stroke signal. This will leave HPCS as the primary high pressure injection system available to maintain Rx level.

Event 8 – HPCS suppression pool leak (Initial Setup - Automatic)

HPCS suppression pool suction line will develop a leak between the suppression pool and its suction isolation valve E22-F015. Suppression pool level will start lowering. EP-3 should be entered and Suppression Pool Makeup (SPMU) should be initiated (may auto initiate). Both divisions of SPMU will receive an initiation signal. fail. SPMU valves E30-F001A is already inop with its breaker open. E30-F002B will experience a stem/disc separation resulting in no water being added to the suppression pool. HPCS room level will start rising and will lead to a HPCS pump trip as water gets in the motor. After the pump trip, HPCS pump room door will fail resulting in a continual drain of the suppression pool into the auxiliary building. With the HPCS pump trip, RPV level will start lowering and will require entry into the Alternate Level Control Leg of EP-2. RCIC should be aligned for injection, CRD maximized for flow, and SLC injected. Emergency depressurization is required prior to suppression pool level falling below 14.5 feet. After emergency depressurization, low pressure ECCS will restore RPV level. CT-1 and CT-2.

GGNS 2-2020 NRC Scenario 4

Termination:

- Once emergency depressurization has been conducted and reactor water level is stabilized above TAF, 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.

GGNS 2-2020 NRC Scenario 4

Critical Task	(CT-1) Emergency Depressurize the RPV prior to Suppression Pool level reaching 14.5 ft.
Event	8
Safety Significance	02-S-01-40 Att. VI, EP Steps SPL-6 through 9. Suppression pool water must be maintained above 14.5 ft. 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 ft., emergency RPV depressurization is required since the RPV is not permitted to remain at pressure if pressure suppression capability is unavailable. Consistent with the definition of “cannot be maintained” a decision that suppression pool water level cannot be maintained above 14.5 ft can be made before level actually reaches this value.
Cueing	Red light indication on at least 7 SRVs

GGNS 2-2020 NRC Scenario 4

Critical Task (CT-2)	(CT-2) After Emergency Depressurization, restore and maintain RPV level above -191” using available injection systems prior to exiting EP-2.
Event	8
Safety Significance	02-S-01-40 Att. IV, EP Step L-14 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. Adequate core cooling is ensured if one of two conditions exists after the RPV is depressurized: • RPV water level can be restored and maintained above the Minimum Steam Cooling RPV Water Level (-191 in.). The core is then cooled by a combination of submergence and steam cooling, even with no core spray flow. • Design core spray flow requirements are satisfied (HPCS or LPCS flow above 7000 gpm and RPV water level above -216 in., the elevation of the top of the jet pumps). The core is then cooled by spray cooling, even if the core remains uncovered.
Cueing	Positive injection flow established and reactor level rising

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

**Per 02-S 01-40, EP-1, Step ED-6: Seven open SRVs is the Minimum Number of SRVs Required for Emergency Depressurization (MNSRED) and is the least number of SRVs which corresponds to a Minimum Steam Cooling Pressure (MSCP) sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow at the corresponding MSCP. The MNSRED is utilized to assure the RPV will depressurize and remain depressurized when emergency depressurization is required. Refer to Appendix A for a detailed discussion of the MNSRED and the MSCP.

SCENARIOS 1 & 3 OMITTED AS THEY WERE UNUSED SPARE SCENARIOS AND WERE PLACED IN THE LICENSEE'S EXAM BANK