Facility: **Grand Gulf Nuclear Station** Date of Exam: **December 2017**

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	**Total**	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	4	N/A			3	4	N/A		3	**20**			
	2	1	2	1				1	1			1	**7**			
	Tier Totals	**4**	**5**	**5**				**4**	**5**			**4**	**27**			
2. Plant Systems	1	2	3	3	3	3	2	1	2	2	3	2	**26**			
	2	2	1	1	1	1	1	1	1	1	1	1	**12**			
	Tier Totals	**4**	**4**	**4**	**4**	**4**	**3**	**2**	**3**	**3**	**4**	**3**	**38**			
3. Generic Knowledge and Abilities Categories					1		2		3		4		**10**	1 / 2	3 / 4	
					3		**3**		**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.
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	K 1	K 2	K 3	A 1	A2	G*	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	X						Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : AK1.02 Power/flow distribution	3.3	1
295003 Partial or Complete Loss of AC / 6			X				Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : AK3.01 Manual and auto bus transfer	3.3	2
295004 Partial or Total Loss of DC Pwr / 6		X					Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: AK2.03 D.C. bus loads	3.3	3
295005 Main Turbine Generator Trip / 3				X			Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP : AA1.05 Reactor/turbine pressure regulating system	3.6	4
295006 SCRAM / 1			X				Knowledge of the reasons for the following responses as they apply to SCRAM : AK3.04 Reactor water level setpoint setdown: Plant-Specific	3.1	5
295016 Control Room Abandonment / 7					X		Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT : AA2.02 Reactor water level	4.2	6
295018 Partial or Total Loss of CCW / 8				X			Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : AA1.02 System loads	3.3	7
295019 Partial or Total Loss of Inst. Air / 8						X	2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	4.2	8
295021 Loss of Shutdown Cooling / 4					X		Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : AA2.02 RHR/shutdown cooling system flow	3.4	9
295023 Refueling Acc / 8		X					Knowledge of the interrelations between REFUELING ACCIDENTS and the following: AK2.06 Containment ventilation: Mark-III	3.4	10
295024 High Drywell Pressure / 5						X	2.2.37 Ability to determine operability and/or availability of safety related equipment.	3.6	11
295025 High Reactor Pressure / 3	X						Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : EK1.03 Safety/relief valve tailpipe temperature/pressure relationships	3.6	12
295026 Suppression Pool High Water Temp. / 5					X		Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: EA2.03 Reactor pressure	3.9	13
295027 High Containment Temperature / 5						X	2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.	3.9	14
295028 High Drywell Temperature / 5		X					Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: EK2.03 Reactor water level indication	3.6	15
295030 Low Suppression Pool Wtr Lvl / 5				X			Ability to operate and/or monitor the following as they apply to LOW SUPPRESSION POOL WATER LEVEL: EA1.03 HPCS: Plant-Specific	3.4	16

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	#
295031 Reactor Low Water Level / 2					X		Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : EA2.04 Adequate core cooling	4.6*	17
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1			X				Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : EK3.01 Recirculation pump trip/runback: Plant-Specific	4.1	18
295038 High Off-site Release Rate / 9							NOT SAMPLED		--
600000 Plant Fire On Site / 8	X						Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: AK1.02 Fire Fighting	2.9	19
700000 Generator Voltage and Electric Grid Disturbances / 6			X				Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AK3.02 Actions contained in abnormal operating procedure for voltage and grid disturbances	3.6	20
K/A Category Totals:	**3**	**3**	**4**	**3**	**4**	**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	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3	X						Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM: AK1.03 Loss of heat sink	3.6	21
295007 High Reactor Pressure / 3							NOT SAMPLED		
295008 High Reactor Water Level / 2							NOT SAMPLED		
295009 Low Reactor Water Level / 2		X					Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: AK2.02 Reactor water level control	3.9	22
295010 High Drywell Pressure / 5							NOT SAMPLED		
295011 High Containment Temp / 5			X				Knowledge of the reasons for the following responses as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY): AK3.01 Increased containment cooling: Mark-III	3.6	23
295012 High Drywell Temperature / 5							NOT SAMPLED		
295013 High Suppression Pool Temp. / 5							NOT SAMPLED		
295014 Inadvertent Reactivity Addition / 1				X			Ability to operate and/or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: AA1.06 Reactor/turbine pressure regulating system	3.3	24
295015 Incomplete SCRAM / 1							NOT SAMPLED		
295017 High Off-site Release Rate / 9							NOT SAMPLED		
295020 Inadvertent Cont. Isolation / 5 & 7					X		Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: AA2.03 Reactor power	3.7	25
295022 Loss of CRD Pumps / 1							NOT SAMPLED		
295029 High Suppression Pool Wtr Lvl / 5						X	2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	4.5	26
295032 High Secondary Containment Area Temperature / 5							NOT SAMPLED		
295033 High Secondary Containment Area Radiation Levels / 9		X					Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following: EK2.01 Area radiation monitoring system	3.8	27
295034 Secondary Containment Ventilation High Radiation / 9							NOT SAMPLED		
295035 Secondary Containment High Differential Pressure / 5							NOT SAMPLED		
295036 Secondary Containment High Sump/Area Water Level / 5							NOT SAMPLED		
500000 High CTMT Hydrogen Conc. / 5							NOT SAMPLED		
K/A Category Point Totals:	1	2	1	1	1	1	**Group Point Total:**		7

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

Plant Systems - **Tier 2/Group 1 (RO)**

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode				X							X	Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: K4.07 Emergency generator load sequencing --------------------------------- 2.4.31 Knowledge of annunciator alarms, indications, or response procedures.	3.7 --- 4.2	28 --- 29
205000 Shutdown Cooling										X		Ability to manually operate and/or monitor in the control room: A4.01 SDC/RHR pumps	3.7	30
206000 HPCI												N/A for GGNS		
20700 Isol Condenser												N/A for GGNS		
209001 LPCS					X				X			Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : K5.01 Indications of pump cavitation ---------------------------------- Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: A3.06 Lights and alarms	2.6 --- 3.6	31 --- 32
209002 HPCS								X				Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.13 Low condensate storage tank level BWR-5,6	3.4	33
211000 SLC	X						X					Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: K1.09 Core spray system: Plant-Specific ------------------------------------- Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: A1.04 Valve operations	3.2* --- 3.6	34 --- 35
212000 RPS						X						Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM : K6.01 A.C. electrical distribution	3.6	36
215003 IRM					X							Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : K5.03 Changing detector position	3.0	37
215004 Source Range Monitor				X								Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: K4.01 Rod withdrawal blocks	3.7	38

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	#
215005 APRM / LPRM			X									Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: K3.05 Reactor power indication	3.8	39
217000 RCIC		X										Knowledge of electrical power supplies to the following: K2.02 RCIC initiation signals (logic)	2.8*	40
218000 ADS	X											Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: K1.05 Remote shutdown system: Plant-Specific	3.9	41
223002 PCIS/Nuclear Steam Supply Shutoff			X									Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: K3.10 Reactor water cleanup	2.9	42
239002 SRVs		X										Knowledge of electrical power supplies to the following: K2.01 SRV solenoids	2.8*	43
259002 Reactor Water Level Control				X								Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: K4.12 Manual and automatic control of the system	3.5	44
261000 SGTS						X						Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : K6.04 Process radiation monitoring	2.9	45
262001 AC Electrical Distribution			X		X							Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following: K3.02 Emergency generators -- Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: K5.01 Principle involved with paralleling two A.C. sources	3.8 ----- 3.1	46 --- 47
262002 UPS (AC/DC)										X		Ability to manually operate and/or monitor in the control room: A4.01 Transfer from alternative source to preferred source	2.8	48
263000 DC Electrical Distribution									X			Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: A3.01 Meters, dials, recorders, alarms, and indicating lights	3.2	49

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
264000 EDGs								X		X		Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 Operating unloaded, lightly loaded, and highly loaded ----------------------------------- Ability to manually operate and/or monitor in the control room: A4.05 Transfer of emergency generator (with load) to grid	3.4 --- 3.6	50 --- 51
300000 Instrument Air		X										Knowledge of electrical power supplies to the following: K2.01 Instrument air compressor	2.8	52
400000 Component Cooling Water											X	2.1.28 Knowledge of the purpose and function of major system components and controls.	4.1	53
K/A Category Point Totals:	**2**	**3**	**3**	**3**	**3**	**2**	**1**	**2**	**2**	**3**	**2**	**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												NOT SAMPLED		
201002 RMCS												N/A for GGNS		
201003 Control Rod and Drive Mechanism												NOT SAMPLED		
201004 RSCS												N/A for GGNS		
201005 RCIS								X				Ability to (a) predict the impacts of the following on the ROD CONTROL AND INFORMATION SYSTEM (RCIS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.04 Withdraw block: BWR-6	3.2	54
201006 RWM												N/A for GGNS		
202001 Recirculation	X											Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION SYSTEM and the following: K1.19 Feedwater flow	3.2	55
202002 Recirculation Flow Control												NOT SAMPLED		
204000 RWCU			X									Knowledge of the effect that a loss or malfunction of the REACTOR WATER CLEANUP SYSTEM will have on following: K3.06 Area radiation levels	2.6	56
214000 RPIS												N/A for GGNS		
215001 Traversing In-Core Probe												NOT SAMPLED		
215002 RBM												N/A for GGNS		
216000 Nuclear Boiler Inst.												NOT SAMPLED		
219000 RHR/LPCI: Torus/Pool Cooling Mode					X							Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: TORUS / SUPPRESSION POOL COOLING MODE: K5.04 Heat exchanger operation	2.9	57
223001 Primary CTMT and Aux.												NOT SAMPLED		
226001 RHR/LPCI: CTMT Spray Mode							X					Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: A1.10 Emergency generator loading	3.0	58
230000 RHR/LPCI: Torus/Pool Spray Mode												N/A for GGNS		
233000 Fuel Pool Cooling/Cleanup												NOT SAMPLED		
234000 Fuel Handling Equipment										X		Ability to manually operate and/or monitor in the control room: A4.01 Neutron monitoring system	3.7	59
239001 Main and Reheat Steam												NOT SAMPLED		
239003 MSIV Leakage Control												NOT SAMPLED		

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

Plant Systems - **Tier 2/Group 2 (RO)**

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
241000 Reactor/Turbine Pressure Regulator									X			Ability to monitor automatic operations of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM including: A3.08 Steam bypass valve operation	3.8	60
245000 Main Turbine Gen. / Aux.						X						Knowledge of the effect that a loss or malfunction of the following will have on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: K6.06 Electrical distribution	3.0	61
256000 Reactor Condensate		X										Knowledge of electrical power supplies to the following: K2.01 System pumps	2.7*	62
259001 Reactor Feedwater	X											Knowledge of the physical connections and/or cause-effect relationships between REACTOR FEEDWATER SYSTEM and the following: K1.08 Reactor water level control system	3.6	63
268000 Radwaste												NOT SAMPLED		
271000 Offgas												NOT SAMPLED		
272000 Radiation Monitoring												NOT SAMPLED		
286000 Fire Protection											X	2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	4.2	64
288000 Plant Ventilation												NOT SAMPLED		
290001 Secondary CTMT												NOT SAMPLED		
290003 Control Room HVAC				X								Knowledge of CONTROL ROOM HVAC design feature(s) and/or interlocks which provide for the following: K4.01 System initiations/reconfiguration: Plant-Specific	3.1	65
290002 Reactor Vessel Internals												NOT SAMPLED		
204000 RWCU												NOT SAMPLED		
K/A Category Point Totals:	**2**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**1**	**Group Point Total:**		**12**

Facility: **Grand Gulf Nuclear Station**							**Date of Exam: December 2017**
Category	K/A #	Topic	RO		SRO-Only		
			IR	#	IR	#	
1. Conduct of Operations	2.1.1	Knowledge of conduct of operations requirements.	3.8	66			
	2.1.5	Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.	2.9*	67			
	2.1.45	Ability to identify and interpret diverse indications to validate the response of another indication.	4.1	68			
	Subtotal			**3**			
2. Equipment Control	2.2.12	Knowledge of surveillance procedures.	3.7	69			
	2.2.14	Knowledge of the process for controlling equipment configuration or status.	3.9	70			
	2.2.43	Knowledge of the process used to track inoperable alarms.	3.0	71			
	Subtotal			**3**			
3. Radiation Control	2.3.13	Knowledge of radiological safety 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, aligning filters, etc.	3.4	72			
	2.3.14	Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.	3.4	73			
	Subtotal			**2**			
4. Emergency Procedures / Plan	2.4.12	Knowledge of general operating crew responsibilities during emergency operations.	4.0	74			
	2.4.34	Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	4.2	75			
	Subtotal			**2**			
Tier 3 Point Total				**10**			

<table>
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6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.
7. The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A catalog, but the topics must be relevant to the applicable evolution or system. Refer to Section D.1.b of ES-401 for the applicable K/As.
8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' 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 (SRO)

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					X		Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: AA2.04 System lineups	3.7	76
295004 Partial or Total Loss of DC Pwr / 6									
295005 Main Turbine Generator Trip / 3					X		Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : AA2.04 Reactor pressure	3.8	77
295006 SCRAM / 1						X	2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	4.2	78
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					X		Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING AA2.03 Reactor water level	3.5	79
295023 Refueling Acc / 8						X	2.4.3 Ability to identify post-accident instrumentation.	3.9	80
295024 High Drywell Pressure / 5									
295025 High Reactor Pressure / 3									
295026 Suppression Pool High Water Temp. / 5									
295027 High Containment Temperature / 5					X		Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) EA2.01 Containment temperature: Mark-III	3.7	81
295028 High Drywell Temperature / 5									
295030 Low Suppression Pool Wtr Lvl / 5									
295031 Reactor Low Water Level / 2									
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1									
295038 High Off-site Release Rate / 9									
600000 Plant Fire On Site / 8						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	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 Emergency and Abnormal Plant Evolutions - **Tier 1/Group 2 (SRO)**	Form ES-401-1

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3									
295007 High Reactor Pressure / 3					X		Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE AA2.02 Reactor power	4.1	83
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					X		Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM AA2.02 Control rod position	4.2	84
295017 High Off-site Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1									
295029 High Suppression Pool Wtr Lvl / 5									
295032 High Secondary Containment Area Temperature / 5									
295033 High Secondary Containment Area Radiation Levels / 9									
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						X	2.4.41 Knowledge of the emergency action level thresholds and classifications.	4.6	85
K/A Category Point Totals:	**0**	**0**	**0**	**0**	**2**	**1**	**Group Point Total:**		**3**

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode											X	G2.1.23: Ability to perform specific system and integrated plant procedures during all modes of plant operations	4.4	86
205000 Shutdown Cooling														
206000 HPCI												N/A for GGNS		
20700 Isol Condenser												N/A for GGNS		
209001 LPCS														
209002 HPCS														
211000 SLC														
212000 RPS								X				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: A2.11 Main steamline isolation valve closure	4.1	87
215003 IRM														
215004 Source Range Monitor														
215005 APRM / LPRM														
217000 RCIC														
218000 ADS								X				Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.05 Loss of A.C. or D.C. power to ADS valves	3.6	88
223002 PCIS/Nuclear Steam Supply Shutoff														
239002 SRVs											X	G2.2.12 Knowledge of Surveillance procedures	4.1	89
259002 Reactor Water Level Control														
261000 SGTS														
262001 AC Electrical Distribution														
262002 UPS (AC/DC)														
263000 DC Electrical Distribution														
264000 EDGs								X				Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.09 Loss of A.C. power	4.1	90
300000 Instrument Air														
400000 Component Cooling 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														
201002 RMCS												N/A for GGNS		
201003 Control Rod and Drive Mechanism														
201004 RSCS												N/A for GGNS		
201005 RCIS														
201006 RWM												N/A for GGNS		
202001 Recirculation								X				Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.12 Loss of reactor feedwater	3.8	91
202002 Recirculation Flow Control														
204000 RWCU														
214000 RPIS												N/A for GGNS		
215001 Traversing In-Core Probe														
215002 RBM												N/A for GGNS		
216000 Nuclear Boiler Inst.														
219000 RHR/LPCI: Torus/Pool Cooling Mode														
223001 Primary CTMT and Aux.														
226001 RHR/LPCI: CTMT Spray Mode														
230000 RHR/LPCI: Torus/Pool Spray Mode												N/A for GGNS		
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment				X								Knowledge of FUEL HANDLING EQUIPMENT design feature(s) and/or interlocks which provide for the following: K4.02 †Prevention of control rod movement during core alterations	4.1	92
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											X	2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.	4.2	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**	**1**	**0**	**0**	**1**	**Group Point Total:**		**3**

Facility: **Grand Gulf Nuclear Station**			Date of Exam: **December 2017**			
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.2	Knowledge of operator responsibilities during all modes of plant operation.			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**

RO EXAM

Tier/Group	Randomly Selected K/A	Reason for Rejection
1 / 1	295016 – AA2.01	AA2. Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: AA2.01 Reactor power At GGNS we are unable to determine Reactor Power external from the main control room. Also, having an ATWS during a Control Room Abandonment is outside our design bases. Randomly selected K/A – AA2.02, Reactor water level.
1 / 1	295027 – 2.1.19	2.1.19 Ability to use plant computers to evaluate system or component status. At GGNS the use of plant computers to evaluate Containment Temperatures is limited, Control Room panel indication is used. Randomly selected K/A – 2.1.25, Ability to interpret reference materials, such as graphs, curves, tables, etc.
1 / 1	295037 – EK3.08	EK3. Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : EK3.08 ATWS circuitry: Plant-Specific GGNS doesn't have an "ATWS Circuitry". Randomly selected K/A – EK3.01, Recirculation pump trip/runback
2 / 1	215004 – K4.02	K4. Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: K4.02 Reactor SCRAM signals The SRMs do not provide Scram signals unless the shorting links are removed. There are no current procedures that will allow the shorting links to be removed at GGNS. Randomly selected K/A – K4.01, Withdrawal Blocks

2 / 1	223002 – K3.20	K3. Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: K3.20 Standby gas treatment system At GGNS the Primary Containment Isolation system has no affect on the Standby gas treatment system. Randomly selected K/A – K3.10, RWCU
2 / 1	262002 – A1.02	A1. Ability to predict and/or monitor changes in parameters associated with operating the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) controls including: A1.02 Motor generator outputs Motor Generators (MGs) are not used for Uninterruptable power at GGNS. Randomly selected K/A – A4.01 A4. Ability to manually operate and/or monitor in the control room: A4.01 Transfer from alternative source to preferred source. A1 had only one other selection, it has a 2.4 importance factor, therefore A4 was randomly selected.
2 / 2	234000 – A4.02	A4. Ability to manually operate and/or monitor in the control room: A4.02 Control rod drive system There is no tie between Fuel Handling Equipment and the CRD system at GGNS. Randomly selected K/A – A4.01, Neutron Monitoring system.

Facility: **GRAND GULF NUCLEAR STATION** Date of Examination: **12/4/2017**

Examination Level: RO ☒ SRO ☐ Operating Test Number: **LOT12-2017**

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R; D	Primary Containment Water Level Determination GJPM-OPS-2017IAR1 K/A 2.1.25: 3.9; 2.1.20: 4.6; 2.4.21: 4.0
Conduct of Operations	R; M	Perform AC/DC Lineup Surveillance GJPM-OPS-2017IAR2 K/A 2.1.31: 4.6; 2.2.12: 3.7; 2.1.20: 4.6
Equipment Control	R; D	Determine Tagging Requirements GJPM-OPS-2017IAR3 K/A 2.2.41: 3.5; 2.2.13: 4.1
Radiation Control		
Emergency Plan	R; N	Emergency Notifications GJPM-OPS-2017IAR4 K/A 2.4.43: 3.2; 2.4.39: 3.9

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

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

Facility: **GRAND GULF NUCLEAR STATION**		Date of Examination: **12/4/2017**
Examination Level: RO ☐ SRO ☒		Operating Test Number: **LOT12-2017**
Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R; M	Perform EOOS Risk Assessment GJPM-OPS-2017IAS1 K/A 2.1.39: 4.3
Conduct of Operations	R; N	Review Completed Surveillance GJPM-OPS-2017IAS2 K/A 2.1.2: 4.4; 2.1.7: 4.7; 2.2.12: 4.1; 2.2.22: 4.7
Equipment Control	R; D; P	Determine Impact on Plant Operations for Failed Relay GJPM-OPS-2017IAS3 K/A 2.2.41: 3.9; 2.2.22: 4.7; 2.2.36: 4.2
Radiation Control	R; M	Authorize Emergency Exposure GJPM-OPS-2017IAS4 K/A 2.3.4: 3.7
Emergency Plan	R; N	Emergency Classification GJPM-OPS-2017IAS5 K/A 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 & Criteria:
(C)ontrol room, (S)imulator, or Class(R)oom
(D)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)
(N)ew or (M)odified from bank (≥ 1)
(P)revious 2 exams (≤ 1; randomly selected)

Facility: **GRAND GULF NUCLEAR STATION**	Date of Examination: **12/04/2017**
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐	Operating Test No.: **LOT12-2017**

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

System / JPM Title	Type Code*	Safety Function
a. 217000 A4.04 (3.6), RCIC Manual Startup, GJPM-OPS-2017IS1	A-D-S	2
b. 239001 A2.11 (4.1), Slow Closing MSIVs, GJPM-OPS-2017IS2	P-A-D-S	3
c. 205000 A4.01 (3.7), Startup Shutdown Cooling A, GJPM-OPS-2017IS3	A-D-L-S	4
d. 264000 A4.04 (3.7), Start, Parallel and Load Div. 1 DG, GJPM-OPS-2017IS4	A-D-S	6
e. 223001 A4.13 (3.4), Startup Hydrogen Recombiner, GJPM-OPS-2017IS5	EN-D-S	5
f. 261000 A4.02 (3.1), Secure Standby Gas Treatment following Automatic Initiation, GJPM-OPS-2017IS6	EN-N-S	9
g. 201005 A2.04 (3.2), Bypass a Control Rod in RACS, GJPM-OPS-2017ICR1	D-C	7
h. 202001 A4.01 (3.7), Shift Reactor Recirc Pumps to Fast Speed, GJPM-OPS-2017IS7	A-D-S	1

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

System / JPM Title	Type Code*	Safety Function
i. 219000 A4.01 (3.8), Align Suppression Pool Cooling from Remote Shutdown Panel, GJPM-OPS-2017IP1	P-A-E-D	5
j. 218000 A2.03 (3.4), Install Nitrogen Bottles on ADS Air Supply, GJPM-OPS-2017IP2	P-E-D-R	3
k. 295037 EA1.03 (4.1), Locally Initiate ATWS ARI, GJPM-OPS-2017IP3	E-N-R	1

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

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

Facility: **GRAND GULF NUCLEAR STATION**	Date of Examination: **12/04/2017**
Exam Level: RO ☐ SRO-I ☒ SRO-U ☐	Operating Test No.: **LOT12-2017**

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

System / JPM Title	Type Code*	Safety Function
a. 217000 A4.04 (3.6), RCIC Manual Startup, GJPM-OPS-2017IS1	A-D-S	2
b. 239001 A2.11 (4.3), Slow Closing MSIVs, GJPM-OPS-2017IS2	P-A-D-S	3
c. 205000 A4.01 (3.7), Startup Shutdown Cooling A, GJPM-OPS-2017IS3	A-D-L-S	4
d. 264000 A4.04 (3.7), Start, Parallel and Load Div. 1 DG, GJPM-OPS-2017IS4	A-D-S	6
e. 223001 A4.13 (3.4), Startup Hydrogen Recombiner, GJPM-OPS-2017IS5	EN-D-S	5
f. 261000 A4.02 (3.1), Secure Standby Gas Treatment following Automatic Initiation, GJPM-OPS-2017IS6	EN-N-S	9
g. 201005 A2.04 (3.2), Bypass a Control Rod in RACS, GJPM-OPS-2017ICR1	D-C	7
h. N/A		

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

System / JPM Title	Type Code*	Safety Function
i. 219000 A4.01 (3.7), Align Suppression Pool Cooling from Remote Shutdown Panel, GJPM-OPS-2017IP1	P-A-E-D	5
j. 218000 A2.03 (3.6), Install Nitrogen Bottles on ADS Air Supply, GJPM-OPS-2017IP2	E-D-R	3
k. 295037 EA1.03 (4.1), Locally Initiate ATWS ARI, GJPM-OPS-2017IP3	E-N-R	1

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

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

Facility: **Grand Gulf Nuclear Station** Date of Exam: 12/2017 Operating Test No.: LOT12/2017

APPLICANT	EVENT TYPE	Scenarios 1			2			3			4			TOTAL	MINIMUM(*)		
		CREW POSITION			CREW POSITION			CREW POSITION			CREW POSITION						
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP		R	I	U
I1 & I2	RX	0			0				1					1	1	1	0
	NOR	1			1				1					3	1	1	1
	I/C	5			5				1					11	4	4	2
	MAJ	1			1				2					4	2	2	1
	TS	3			2									5	0	2	2
I3	RX	0					0		1					1	1	1	0
	NOR	1					1		1					3	1	1	1
	I/C	5					4		1					10	4	4	2
	MAJ	1					1		2					4	2	2	1
	TS	3												3	0	2	2
I4	RX		1		0			0						1	1	1	0
	NOR		0		1			1						2	1	1	1
	I/C		2		5			3						10	4	4	2
	MAJ		1		1			2						4	2	2	1
	TS				2			2						4	0	2	2
R1 R3	RX		1				0							1	1	1	0
	NOR		0				1							1	1	1	1
	I/C		2				4							6	4	4	2
	MAJ		1				1							2	2	2	1
	TS														0	2	2
R2 R4 R5	RX			0		1				0				1	1	1	0
	NOR			1		0				1				2	1	1	1
	I/C			3		2				3				8	4	4	2
	MAJ			1		1				2				4	2	2	1
	TS														0	2	2
B/U	RX										0	0	0		1	1	0
	NOR										1	1	0		1	1	1
	I/C										6	3	3		4	4	2
	MAJ										1	1	1		2	2	1
	TS										2				0	2	2

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.

Facility: Grand Gulf Nuclear Station Scenario No.: 1 Op-Test No.: GGNS 12-2017

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

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

1. Start RHR A in Suppression Pool Cooling.
2. Respond to a LPCS Jockey Pump trip.
3. Respond to a RWCU Division 1 isolation signal with failure of G33-F250 to isolate due to power loss.
4. Respond to a Reactor Feed Pump A Speed Controller failure.
5. Respond to a Low Pressure Feedwater 4A tube leak with a failure to isolate.
6. Respond to a RCIC Steam Line break with a failure to isolate.
7. Respond to a low power ATWS (<5%) with failed fuel.
8. Respond to a HPCS Service Water Pump trip.

Initial Conditions: Plant is operating at approximately 100% power. Average Suppression Pool temperature is 85°F due to weeping SRVs.

Inoperable Equipment: None

Turnover:

The plant is at 100% power.

Planned activities for this shift are:

- Start RHR A in Suppression Pool Cooling to lower Suppression Pool temperature to 78°F to 80°F.

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 60 minutes

Event No.	Malf. No.	Event Type †	Event Description
1	N/A	TS (CRS) N (BOP,CRS)	Start RHR A in Suppression Pool Cooling.
2	e21645	TS (CRS) C (BOP,CRS)	Respond to a LPCS Jockey Pump trip.
3	tte31n043a g33f250	TS (CRS) I(BOP,CRS) A (CREW)	Respond to a Division 1 RWCU Isolation signal with G33-F250 power loss.
4	fw272a	C(ATC,CRS) A (CREW)	Respond to a Reactor Feed Pump A Speed Controller failure.
5	n19f042a_f n19f040a_f fw232j	C (BOP,CRS) R (ATC) A (CREW)	Respond to a LP FW HTR 4A tube leak with a failure to isolate.
6	e51050 e51187a e51187b	M (CREW)	Respond to RCIC steam line break with a failure to isolate.
7	c11164 rr071	M (CREW)	ATWS < 5% power with Fuel Failure *** (CT-1) When control rods fail to scram, crew inserts control rods before exiting EP-2A. (All control rods do not have to be fully inserted to satisfy this critical task; this only requires that the crew is making progress to achieving all rods in by fully inserting at least 5 control rod gangs using RC&IS.)** *** (CT-2) When two areas exceed their max safe radiation levels, emergency depressurize the RPV before exiting EP-4.** *** (CT-3) When emergency depressurization is required, crew terminates and prevents all RPV injection, except RCIC, CRD, and Boron, prior to emergency depressurizing the RPV.**
8	r21133b	C (ATC)	Respond to a HPCS Service Water Pump trip.

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

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

CREW notation for Abnormal (A) and Major (M) events denotes ATC, BOP, and CRS are credited.

Quantitative Attributes Table

Attribute	E3-304-1 Target	Actual	Description
Malfunctions after EOP entry	1-2	1	• HPCS Service Water Pump trip
Abnormal Events	2-4	3	• Division 1 RWCU Isolation signal with G33-F250 power loss • Reactor Feed Pump A Speed Controller failure • LP FW HTR 4A leak with a failure to isolate
Major Transients	1-2	2	• RCIC steam line break with a failure to isolate • ATWS < 5% power/Fuel Failure
EOP entries requiring substantive action	1-2	2	• EP-4 • EP-2
EOP contingencies requiring substantive action	0-2	2	• EP-2A ATWS • EP-2A Emergency Depressurization
EOP based Critical Tasks	2-3	3	• **(CT-1) When control rods fail to scram, crew inserts control rods before exiting EP-2A. (All control rods do not have to be fully inserted to satisfy this critical task; this only requires that the crew is making progress to achieving all rods in by fully inserting at least 5 control rod gangs using RC&IS.)** • **(CT-2) When two areas exceed their max safe radiation levels, emergency depressurize the RPV before exiting EP-4** • **(CT-3) When emergency depressurization is required, crew terminates and prevents all RPV injection, except RCIC, CRD, and Boron, prior to emergency depressurizing the RPV**
Normal Events	N/A	1	• Start RHR A in Suppression Pool Cooling
Reactivity Manipulations	N/A	1	• Lower core flow to 70 mlbm/hr using Recirculation Flow Control Valves
Instrument / Component failures	N/A	5	• LPCS Jockey Pump trip • Division 1 RWCU Isolation signal with G33-F250 power loss • Reactor Feed Pump A Speed Controller failure • LP FW HRT 4A tube leak with failure to isolate • HPCS Service Water Pump trip
Total Malfunctions	N/A	7	• LPCS Jockey Pump trip • Division 1 RWCU Isolation signal with G33-F250 power loss • Reactor Feed Pump A Speed Controller failure • LP FW HRT 4A tube leak with failure to isolate • RCIC steam line break with a failure to isolate • ATWS < 5% power/Fuel Failure • HPCS Service Water Pump trip

<u>Top 10 systems and operator actions important to risk that are tested:</u>

ADS (Event 7)
RPS (Event 7)
Condensate (Event 4)
Failure to manually depressurize with ADS/SRVs (Event 7)

SCENARIO ACTIVITIES:

Plant is operating at 100% power. Average Suppression Pool temperature is 85°F due to weeping SRVs.

Event 1

After the crew assumes the shift, the BOP will start RHR A in Suppression Pool Cooling mode. The CRS will enter Tech Spec 3.5.1 Condition A when E12-F024A, RHR A Test Return valve, is opened.

Event 2 **(Triggered by Lead Examiner)**

When RHR A is aligned in Suppression Pool Cooling mode and Tech Specs are addressed, the LPCS Jockey Pump will trip. The CRS will enter TS 3.5.1 Condition C and may direct BOP to place RHR A in standby. If E12-F024A, RHR A Test Return valve, is closed, CRS will exit Tech Spec 3.5.1 Condition C and determine Tech Spec 3.5.1 Condition A is still applicable. CRS will enter Tech Spec 3.3.6.4, Condition A and C, 3.3.3.1, Condition A, 3.3.3.2, Condition A and TR3.6.2.2, Condition A.

Event 3 **(Triggered by Lead Examiner)**

After RHR A has been secured from Suppression Pool Cooling and Tech Specs have been addressed, a Division 1 isolation of RWCU will occur due to a failed temperature instrument. G33-F250, RWCU SPLY TO RWCU HXS, will lose power and fail to isolate. CRS will enter 05-1-02-III-5, Automatic Isolations ONEP and direct BOP to close G33-F251, RWCU SPLY TO RWCU HXS. The CRS will enter Tech Specs 3.3.6.1 Condition A and 3.6.5.3 Condition A.

Event 4 **(Triggered by Lead Examiner)**

After G33-F251 has been closed and Tech Specs are addressed, the Reactor Feed Pump A Speed Controller will begin failing low. ATC will take manual control of the controller and balance Reactor Feed Pump controller outputs and stabilize RPV level. CRS will enter 05-1-02-V-7, Feedwater System Malfunctions ONEP.

Event 5 **(Triggered by Lead Examiner)**

When Reactor Feed Pump controller outputs have been balanced and RPV level has stabilized, a tube leak in Low Pressure Feedwater Heater 4A will occur. The heater isolation valves, N19-F040A and N19-F042A will fail to automatically close on the HI-HI level. The BOP will manually close the isolation valves to isolate the heater. The CRS enter 05-1-02-V-5, Loss of Feedwater Heating ONEP, and direct the ATC to lower core flow to 70 mlbm/hr using Recirc Flow Control Valves in slow detent. The CRS will enter 05-1-02-III-3, Reduction in Recirculation System Flow Rate ONEP.

Event 6 **(Triggered by Lead Examiner)**

After core flow has been lowered and RPV power, pressure and level are stable, a RCIC steam line break will occur. BOP will attempt to isolate RCIC by closing the RCIC Steam Isolation valves. RCIC steam isolation valves will lose power. CRS will enter EP-4 and direct the ATC to manually scram the reactor.

Event 7 **(Triggered automatically)**

When the reactor is scrammed, an ATWS occurs due to a hydraulic block of both scram discharge volumes with failed fuel, and EP-2A is entered via EP-2. Reactor power is below 5% rated thermal power. The crew will install the necessary attachments to bypass RPS and RC&IS interlocks and insert controls rods manually via RC&IS **(CT-1)**. RPV level will be maintained in the normal band of +11.4" to +53.5" narrow range. Bypass valves will control reactor pressure during this event. Feedwater is available for RPV level control. The crew may decide to lower RPV pressure to reduce the driving head of the leak using manual Bypass Valves control.

When two areas (RCIC Room and SBGTS) exceed the max safe radiation levels of EP-4, the CRS will direct the ATC and BOP to terminate and prevent all injection into the RPV (except RCIC, CRD and BORON) **(CT-3)** and emergency depressurize the RPV **(CT-2)**. When RVP pressure has lowered to Minimum Steam Cooling Pressure of 206 psig, the CRS will direct the crew to slowly commence injection into the RPV with available systems to restore and maintain RPV level between 11.4" and 53.5" narrow range.

After the crew have emergency depressurized the RPV and inserted at least 5 gangs of control rods, or at the discretion of the Lead Examiner, the control rods are allowed to be fully inserted with the next scram. The CRS transitions from EP-2A to EP-2 and RPV level restoration is directed.

Event 8 **(Triggered automatically)**

When High Pressure Core Spray is initiated, the HPCS Service Water Pump will trip. The crew will secure the High Pressure Core Spray Diesel Generator due to lack of cooling water.

The exercise ends when controls rods are inserted and RPV water level is being maintained between +11.4 inches and +53.5 inches narrow range.

Critical Task	**(CT-1) When control rods fail to scram, crew inserts control rods before exiting EP-2A. (All control rods do not have to be fully inserted to satisfy this critical task; this only requires that the crew is making progress to achieving all rods in by fully inserting at least 5 control rod gangs using RC&IS.)**	**(CT-2) When two areas exceed their max safe radiation levels, emergency depressurize the RPV before exiting EP-4.**
EVENT	7	7
Safety Significance	Failure to effect shutdown of the reactor when a RPS setting 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.	If secondary containment radiation levels continue to increase and exceed their maximum safe operating values in more than one area, the RPV must be depressurized. RPV depressurization places the primary system in its lowest possible energy state, rejects heat to the suppression pool in preference to outside the containment, and reduces the driving head and flow of primary systems that are unisolated and discharging into the secondary containment.
Cueing	Manual scram is initiated and numerous control rods indicate beyond position 02. Reactor power indicating > 0% rated thermal power.	EP-4 max safe indication on PDS computer points. Radiation levels exceeding max safe values on area radiation instrumentation on panel P844.
Performance Indicator	Operator selects control rod gangs by depressing the respective pushbuttons on panel P680 and inserts the rods by depressing the IN-TIMER SKIP pushbutton.	Manipulation of seven of the eight ADS/SRVs on panel P601: B21-F041K B21-F047L B21-F041F B21-F047A B21-F051C B21-F041D B21-F051A B21-F051B
Performance Feedback	Operator selecting and inserting control rods indicated by rod position decreasing to 00 for selected rods on panel P680	Crew will observe ADS/SRV light indication go from green to red and reactor pressure lowering on SPDS and panel P601 indications.
Justification for the chosen performance limit	There is no time limit for effecting complete reactor shutdown via control rod insertion. For the timeframe of this scenario, containment limits are not challenged and power oscillations are not experienced. However, if the failure to scram EP were to be exited, other procedures would not provide the guidance necessary to achieve reactor shutdown. Before exiting EP-2A ensures guidance to effect reactor shutdown is not removed.	There is no time frame for performing the emergency depressurization of the RPV when two area radiation levels exceed their max safe values. However, if the emergency depressurization is not performed before EP-4 is exited, other procedures would not provide the guidance necessary to direct the depressurization. Before exiting EP-4 ensures this guidance to emergency depressurize the RPV is not removed.
BWR Owners Group Appendix	App. B, step RC/Q-7	App. B, step SC/R-2.2

Critical Task	**(CT-3) When emergency depressurization is required, crew terminates and prevents all RPV injection, except RCIC, CRD, and Boron, prior to emergency depressurizing the RPV.**	
EVENT	7	
Safety Significance	Injection into the RPV is terminated and prevented while emergency RPV depressurization proceeds, in order to prevent uncontrolled injection of large amounts of cold water as RPV pressure decreases below the shutoff head of operating system pumps. Injection from boron injection systems and CRD need not be terminated since the flowrates are relatively small and the systems may be needed to shut down the reactor. RCIC injection need not be terminated since its flowrate is also relatively small, turbine operation helps to depressurize the RPV, and RPV depressurization is not expected to result in significant flow variations.	
Cueing	EP-4 indications on PDS computer points in more than one area. Radiation levels exceeding max safe values on more than one area radiation instruments on panel P844. Manual scram is initiated and numerous control rods indicate beyond position 02.	
Performance Indicator	Operator manipulates Div 1 and Div 2 ECCS and HPCS manual initiation switches and associated pump and injection valve handswitches on panel P601. Operator manipulates Master Level Controller or Startup Level Controller in MANUAL and lowers output to 0%. Operator ensures N21-F009A and B and N21-F040 closed.	
Performance Feedback	Green light on and red light extinguished on ECCS pump and injection valve handswitches on panel P601. ECCS pump and valve overridden annunciators on panel P601. Feedwater flow indicating 0 mlbm/hr on panel P680 instruments. Master Level Controller/Startup Level Controller output indicating 0%. Green light on and red light extinguished on N21-F009A, F009B and F040 handswitches and CLOSE pushbuttons depressed on panel P680.	
Justification for the chosen performance limit	Injection into the RPV is terminated and prevented while emergency RPV depressurization proceeds, in order to prevent uncontrolled injection of large amounts of cold water as RPV pressure decreases below the shutoff head of operating system pumps. Performance of this task before emergency depressurizing the RPV ensures that RPV injection sources are secured prior to RPV pressure lowers below the shutoff head of the associated pumps.	
BWR Owners Group Appendix	App. B, Contingency 5, step C5-5.1	

Facility: Grand Gulf Nuclear Station Scenario No.: 2 Op-Test No.: GGNS 12-2017

Examiners: ______________ Operators: ______________

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

1. Transfer ESF Bus 17AC from ESF Transformer 21 to ESF Transformer 12.
2. Respond to an ESF Transformer 12 Lockout with failure of HPCS Diesel Generator to automatically start.
3. Respond to a CRD Pump B trip.
4. Respond to a control rod drifting in.
5. Respond to a stuck control rod.
6. Respond to a second control rod drifting in.
7. Respond to a Hydraulic Block ATWS with power > 5% RTP
8. Respond to a Reactor Feed Pump trip.

Initial Conditions: Plant is operating at 100% power.

Inoperable Equipment: None

Turnover:

The plant is at 100% power.

Planned activities for this shift are:

- Transfer Bus 17AC from ESF Transformer 21 to ESF Transformer 12 in preparation for red-tagging breaker 152-1705, 17AC FFDR FM ESF 21, for preventative maintenance

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 60 minutes

Event No.	Malf. No.	Event Type †	Event Description
1	N/A	N (BOP,CRS)	Transfer ESF Bus 17AC from ESF Transformer 21 to ESF Transformer 12.
2	r21134h n41141c	TS (CRS) C (BOP,CRS) A (CREW)	Respond to an ESF Transformer 12 Lockout with a failure of HPCS Diesel Generator to auto start.
3	c11028b	C(BOP,CRS) A (CREW)	Respond to a CRD Pump B trip.
4	z022021_24_53	C(ATC,CRS) A(CREW)	Respond to Control Rod 24-53 drifting in.
5	z022022_24_53	R (ATC) C(BOP,CRS) A(CREW) TS (CRS)	Respond to Control Rod 24-53 stuck at position 32.
6	z021021_32_37	C (ATC,CRS)	Respond to Control Rod 32-27 drifting in. Reactor scram due to two controls drifting in.
7	c11164	M(CREW)	Respond to a Hydraulic Block ATWS > 5% RTP (EP-2, 2A) * **(CT-1) When control rods fail to scram, crew injects SLC and/or inserts control rods before exiting EP-2A. (All control rods do not have to be fully inserted to satisfy this critical task; this only requires that the crew is making progress to achieving all rods in by fully inserting at least 5 control rod gangs using RC&IS.)** * **(CT-2) During failure to scram conditions with power > 5% RTP, terminate and prevent all injection from all sources (except boron, CRD, and RCIC) as necessary to lower RPV level to below -70" wide range and control between -70" wide range to -191" CFZ (MSCWL) prior to exiting EP-2A.**
8	fw123a(b)	C(BOP,CRS)	Respond to a Reactor Feedwater Pump trip. ***(CT-3) Restores injection using Condensate/Feedwater to restore/maintain RPV level above -191" CFZ before exiting EP-2A.**
†	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec		
*	**Critical Task** (As defined in NUREG 1021 Appendix D)		

CREW notation for Abnormal (A) and Major (M) events denotes ATC, BOP, and CRS are credited.

Quantitative Attributes Table

Attribute	E3-304-1 Target	Actual	Description
Malfunctions after EOP entry	1-2	1	• Reactor Feed Water Pump trip
Abnormal Events	2-4	4	• ESF Transformer 12 lockout with a failure of HPCS Diesel Generator to automatically start • CRD Pump B trip • Control Rod 24-53 drifting in • Control Rod 24-53 stuck at position 32
Major Transients	1-2	1	• Hydraulic block ATWS with power > 5% RTP
EOP entries requiring substantive action	1-2	2	• EP-2 • EP-3
EOP contingencies requiring substantive action	0-2	1	• EP-2A ATWS
EOP based Critical Tasks	2-3	3	• **(CT-1) When control rods fail to scram, crew injects SLC and/or inserts control rods before exiting EP-2A. (All control rods do not have to be fully inserted to satisfy this critical task; this only requires that the crew is making progress to achieving all rods in by fully inserting at least 5 control rod gangs using RC&IS.)** • **(CT-2) During failure to scram conditions with power > 5% RTP, terminate and prevent all injection from all sources (except boron, CRD, and RCIC) as necessary to lower RPV level to below -70” wide range and control between -70” wide range to -191” CFZ (MSCWL) prior to exiting EP-2A.** • **(CT-3) Restores injection using Condensate/Feedwater to restore/maintain RPV level above -191” CFZ before exiting EP-2A.**
Normal Events	N/A	1	• Transfer ESF Bus 17AC from ESF Transformer 21 to ESF Transformer 12
Reactivity Manipulations	N/A	1	• Lower core flow to 70 mlbm/hr using Reactor Recirc Flow Control Valves
Instrument / Component failures	N/A	6	• ESF Transformer 12 lockout with a failure of HPCS Diesel Generator to automatically start • CRD Pump B trip • Control Rod 24-53 drifting in • Control Rod 24-53 stuck • Control Rod 32-37 drifting in • Reactor Feed Water Pump A(B) trip
Total Malfunctions	N/A	7	• ESF Transformer 12 lockout with a failure of HPCS Diesel Generator to automatically start • CRD Pump B trip • Control Rod 24-53 drifting in • Control Rod 24-53 stuck • Control Rod 32-37 drifting in • Hydraulic block ATWS with power > 5% RTP • Reactor Feed Water Pump A(B) trip

Top 10 systems and operator actions important to risk that are tested:

RPS (Event 6)
ESF Power (Event 2)
Condensate (Event 8)
Failure to align alternate power to 4.16 KV or 6.9 KV buses (Event 2)

SCENARIO ACTIVITIES:

The plant is operating at 100% power.

Event 1

After the crew assumes the shift, the BOP will transfer ESF Bus 17AC from ESF Transformer 21 to ESF Transformer 12 per System Operating Instruction 04-1-01-R21-17, ESF BUS 17AC, Section 4.2.

Event 2

One minute after ESF Bus 17AC has been transferred, ESF Transformer 12 will lockout due to sudden pressure, causing a loss of power to ESF Bus 17AC. The HPCS Diesel Generator will fail to start. The BOP will recognize the failure of the HPCS Diesel Generator to auto start and restore Bus 17AC power from ESF Transformer 21 per 05-1-02-I-4, Loss of AC Power ONEP. The CRS will enter 05-1-02-I-4, Loss of AC Power ONEP, and TS 3.8.1.B for the HPCS Diesel Generator inoperable.

Event 3 **(Triggered by Lead Examiner)**

After Bus 17AC power has been restored and Tech Specs have been addressed, the ‘B’ CRD pump will trip. The BOP will start the ‘A’ CRD pump per 05-1-02-IV-1, CRD Malfunctions ONEP. The CRS will enter 05-1-02-IV-1, CRD Malfunctions ONEP.

Event 4 **(automatically triggered)**

When the ‘A’ CRD pump is started, Control Rod 24-53 will begin drifting in. The ATC will select Control Rod 24-53 and apply a continuous insert signal per 05-1-02-IV-1, CRD Malfunctions ONEP. The CRS will re-enter 05-1-02-IV-1, CRD Malfunctions ONEP.

Event 5 **(automatically triggered)**

When Control Rod 24-53 reaches position 32, it will become stuck. The ATC will recognize and report Control Rod 28-53 has stopped inserting. The ATC will lower core flow to 70 mlbm/hr using Recirc Flow Control Valves in fast detent per 05-1-02-IV-1, CRD Malfunctions ONEP. The CRS will direct actions to raise CRD drive water pressure and attempt to insert Control Rod 24-53. When CRD Drive water pressure is raised to greater than 325 psig, Event 6 will automatically be triggered.

NOTE: Due to the amount of time required to complete the actions associated with the stuck control rod, a follow up question should be asked concerning the stuck rod and Technical Specification requirements (TS 3.1.3, Condition A).

Event 6 **(automatically triggered)**

When CRD Drive Water pressure is raised above 325 psig, Control Rod 32-37 will begin to drift in. The ATC will insert a manual scram per 05-1-02-IV-1, CRD Malfunctions ONEP.

Event 7 **(No trigger required)**

When the reactor is scrammed, an ATWS occurs due to a hydraulic block of both scram discharge volumes, and EP-2A is entered via EP-2. Reactor power will be above 5% RTP. The crew will inject SLC and install the necessary attachments to bypass RPS and RC&IS interlocks and insert controls rods manually via RC&IS **(CT-1)**. 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 and lower reactor power **(CT-2)**. Bypass valves will control reactor pressure during this event. Feedwater is available for RPV level control.

Event 8 **(Triggered when reactor level lowers below -70 inches wide range)**

When reactor level lowers below -70 inches wide range, the running Reactor Feed Pump will trip. The BOP will restore Feedwater flow by starting the standby Reactor Feed Pump **(CT-3)**. An alternate success path would be the CRS the ATC to lower reactor pressure to 450 to 600 psig to allow RPV injection with the Condensate Booster Pumps **(CT-3)**.

After the crew has fully inserted 5 control rod gangs or at the direction of the Lead Examiner, the control rods are allowed to be fully inserted with the next scram. The CRS transitions from EP-2A to EP-2, SLC injection is stopped and RPV level restoration is directed.

The exercise ends when controls rods are inserted and RPV water level is being maintained between +11.4 inches and +53.5 inches narrow range.

Critical Task	**(CT-1) When control rods fail to scram, crew injects SLC and/or inserts control rods before exiting EP-2A. (All control rods do not have to be fully inserted to satisfy this critical task; this only requires that the crew is making progress to achieving all rods in by fully inserting at least 5 control rod gangs using RC&IS.)**	**(CT-2) During failure to scram conditions with power > 5%, terminate and prevent all injection from all sources (except boron, CRD, and RCIC) as necessary to lower RPV level to below -70" wide range to -191" CFZ (MSCWL) prior to exiting EP-2A.**
EVENT	7	7
Safety Significance	Failure to effect shutdown of the reactor when a RPS setting 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.	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	Manual scram is initiated and numerous control rods indicate beyond position 02. Reactor power indicating > 5% RTP on APRMs on panel P680. APRM downscale lights on panel P680APRM extinguished.	Manual scram is initiated 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.
Performance Indicator	Operator manipulates key-locked switches for SLC Pump A and B to START on panel P601. Operator selects control rod gangs by depressing the respective pushbuttons on panel P680 and inserts the rods by depressing the IN-TIMER SKIP pushbutton. Operator resets reactor scram signal with key-locked switches on panel P680 and inserts manual reactor scram using scram pushbuttons on panel P680.	Operator manipulates the Master Level Controller in MANUAL on panel P680 and lowers output to 0% to stop feedwater injection until RPV water level lowers below -70" wide range. Operator manually initiates High Pressure Core Spray and Division 1 and Division 2 ECCS on panel P601, then stops the respective pumps and overrides the associated injection valves closed using their respective handswitches on panel P601.
Performance Feedback	SLC A and B red lights illuminate, SLC discharge pressure rising, and SLC tank level lowering on panel P601. Operator selecting and inserting control rods indicated by rod position decreasing to 00 for selected rods on panel P680.	Feedwater flow indication on panel P680 and SPDS indicate zero. Master Level Controller output indicates 0% on panel P680. High Pressure Core Spray, RHR A, RHR B, and RHR C pump and injection valve override annunciators illuminated on panel P601.

Justification for the chosen performance limit	There is no time limit for effecting complete reactor shutdown via boron injection or control rod insertion. For the timeframe of this scenario, containment limits are not challenged and power oscillations are not experienced. However, if the failure to scram EP were to be exited, other procedures would not provide the guidance necessary to achieve reactor shutdown. Before exiting EP-2A ensures guidance to effect reactor shutdown is not removed.	Applicability for this CT is during EP-2A conditions where it is necessary to lower level to control power with no high energy input into the primary containment. There is no time limit for this lowering level, but it establishes margin to conditions where fuel damaging power oscillations may theoretically occur. Before exiting EP-2A was chosen because other procedures would not provide the guidance necessary to establish margin for power oscillation mitigation. Before exiting EP-2A ensures guidance to effect this control is not removed. NOTE – This critical task must be evaluated carefully based on the level changes. If power is reduced significantly below 5%, reactor water level may continue to rise above -70” with only CRD and SLC while driving control rods. This would not result in an UNSAT on this critical task.
BWR Owners Group Appendix	App. B, step RC/Q6, RC/Q-7	App. B, Contingency #5 Step C5-4

Critical Task	**(CT-3) Restores injection using Condensate/Feedwater to restore/maintain RPV level above -191" CFZ before exiting EP-2A.**	
EVENT	8	
Safety Significance	If RPV water level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level (-191" CFZ), emergency RPV depressurization is performed to maximize injection flow. Emergency depressurization is undesirable under ATWS conditions since the core response is difficult to predict and the risk of power excursions is increased.	
Cueing	Reactor Feed Pump trip annunciators and Feedwater flow and RPV level lowering on indicators on panel P680 and on PDS and SPDS.	
Performance Indicator	Operator manipulates switches on panel P680 panel to start the standby Reactor Feed Pump Alternately, operator lowers RPV pressure using Bypass Valves or SRVs to allow injection with Condensate Booster Pumps.	
Performance Feedback	Feedwater flow and RPV level rising on panel P680 and PDS and SPDS.	
Justification for the chosen performance limit	The Minimum Steam Cooling RPV Water Level (-191" CFZ) 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. Maintaining RPV water level above the Minimum Steam Cooling RPV Water Level thus ensures that the core remains adequately cooled.	
BWR Owners Group Appendix	App. B, Contingency #5 Step C5-4	

Facility: Grand Gulf Nuclear Station Scenario No.: 3 Op-Test No.: GGNS 12-2017

Examiners: ____________ Operators: ____________

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

1. Transfer RPS Bus B from normal to alternate power supply.
2. Respond to a Division 2 Diesel Generator lube oil leak.
3. Respond to a Division 2 ECCS initiation on spurious RPV low level signal.
4. Respond to a loss of condenser vacuum.
5. Respond to a loss of Offsite Power / LOCA
6. Respond to a failure of Division 1 ECCS to automatically initiate.

Initial Conditions: Plant is operating at 100% power.

Inoperable Equipment: TBCW Pump C is tagged out for motor oil replacement.

Turnover:

The plant is at 100% power.

Planned activities for this shift are:

- Transfer RPS Bus B from normal to alternate power supply

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 60 minutes

Event No.	Malf. No.	Event Type †	Event Description
1	N/A	N (BOP,ATC,CRS)	Transfer RPS Bus B from normal to alternate power supply.
2	p864_2a_d_2	TS (CRS) C (BOP,CRS)	Respond to a Division 2 Diesel Generator lube oil leak.
3	ltb21n091b ltb21n091f	I (ATC,BOP,CRS) A(CREW) TS (CRS)	Respond to Division 2 ECCS on spurious RPV low level signal.
4	fw163c	R (ATC,CRS) A(CREW)	Respond to loss of condenser vacuum.
5	r21135 rr063b	M(CREW)	Respond to a LOP/LOCA (EP-2, 3) * **(CT-1) Inhibit ADS prior to automatic ADS valve opening during a LOCA** * **(CT-2) When RPV level lowers to -160” wide range and cannot be maintained above -191” CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191” CFZ. (Momentary shrink below -191” due to automatic SRV closure does not constitute failure of this critical task).**
6	rr040a rr041a	C(BOP,CRS)	Respond to a failure of Division 1 ECCS to automatically initiate. * **(CT-3) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually initiates ECCS systems for injection prior to RPV pressure lowering below 300 psig.**
†	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec		
*	**Critical Task** (As defined in NUREG 1021 Appendix D)		

CREW notation for Abnormal (A) and Major (M) events denotes ATC, BOP, and CRS are credited.

Quantitative Attributes Table

Attribute	E3-304-1 Target	Actual	Description
Malfunctions after EOP entry	1-2	1	• Failure of Division 1 ECCS to automatically initiate
Abnormal Events	2-4	2	• Spurious Division 2 ECCS initiation • Loss of condenser vacuum
Major Transients	1-2	2	• LOP • LOCA
EOP entries requiring substantive action	1-2	2	• EP-2 • EP-3
EOP contingencies requiring substantive action	0-2	2	• EP-2 Alternate Level Control • EP-2 Emergency Depressurization
EOP based Critical Tasks	2-3	3	• **(CT-1) Inhibit ADS prior to automatic ADS valve opening during a LOCA** • **(CT-2) When RPV level lowers to -160" wide range and cannot be maintained above -191" CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191" CFZ. (Momentary shrink below -191" due to automatic SRV closure does not constitute failure of this critical task)** • **(CT-3) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually initiates ECCS systems for injection prior to RPV pressure lowering below 300 psig**
Normal Events	N/A	1	• Transfer RPS Bus B from normal to alternate power supply
Reactivity Manipulations	N/A	1	• Lower core flow to 70 mlbm using Reactor Recirc Flow Control Valves
Instrument / Component failures	N/A	4	• Division 2 Diesel Generator lube oil leak • Spurious Division 2 ECCS initiation • Loss of vacuum • Failure of Division 1 ECCS to automatically initiate
Total Malfunctions	N/A	5	• Division 2 Diesel Generator lube oil leak • Spurious Division 2 ECCS initiation • Loss of vacuum • LOP/LOCA • Failure of Division 1 ECCS to automatically initiate

Top 10 systems and operator actions important to risk that are tested:

Div 1 & 2 EDGs (Event 2)
ADS (Event 5)
Offsite Power (Event 5)
Failure to manually depressurize with ADS/SRVs (Event 5)

SCENARIO ACTIVITIES:

The plant is operating at 100% power. TBCW Pump C is tagged out of service.

Event 1

After the crew assumes the shift, the BOP will transfer RPS Bus B from normal to alternate power supply per System Operating Instruction 04-1-01-C71-1, Reactor Protection System, Section 5.1. The ATC will reset the half-scram.

Event 2 **(Triggered by Lead Examiner)**

After RPS Bus B is transferred to alternate power, annunciator "DIV 2 DSL GEN TROUBLE" will alarm. BOP will dispatch plant operator to investigate. After 3 minutes, the plant operator will report lube oil spraying out from the Division 2 Lube Oil Circulating Pump discharge piping and lube oil sump level is 20" below the top of the sump, which is less than 350 gallons. The BOP will place Division 2 Diesel Generator in the MAINTENANCE Mode. The CRS will enter LCO 3.8.1.B and LCO 3.8.3.E.

Event 3 **(Triggered by Lead Examiner)**

When Tech Specs have been addressed, a spurious Division 2 ECCS initiation on low RPV level will occur. The BOP will verify the initiation is spurious by two independent means and recover from the Division 2 ECCS initiation using 04-1-01-E12-1, Residual Heat Removal System SOI, Attachment 9 hardcard. The ATC will recognize the Division 2 half-scram due to RPS Bus B loss of power. CRS will enter 05-1-02-III-2, Loss of One or Both RPS Buses. BOP will restore RPS Bus B to normal power supply and the ATC will reset the Division 2 half-scram. The CRS will enter LCO 3.3.5.1.B.

Event 4 **(Triggered by Lead Examiner)**

When Division 2 ECCS initiation has been reset, systems have been secured and Tech Specs have been addressed, a main condenser leak will result in a slow loss of condenser vacuum. The CRS will enter 05-1-02-V-8, Loss of Condenser Vacuum ONEP. The ATC will lower core flow to 70 mlbm/hr using Recirc Flow Control Valves in fast detent. When condenser vacuum continues to lower, the ATC will insert a manual scram.

Event 5 **(No trigger required)**

When the reactor is scrammed, a total loss of offsite power occurs, followed by a small recirculation pipe break after 5 minutes. HPCS will trip when it is initiated. The CRS enters EP-2 and EP-3. RPV level will lower due to the leak being greater than the capacity of RCIC. When the CRS determines there are insufficient high pressure injection sources to maintain RPV level above -160" wide range, enters Alternate Level Control contingency of EP-2. Crew will inhibit ADS to prevent automatic operation **(CT-1)**. When RPV level lowers to -160" wide range, the crew will emergency depressurize the RPV using ADS/SRVs **(CT-2)** and restore RPV level with Division 1 ECCS systems.

Event 6 **(No Trigger required)**

Division 1 ECCS will fail to automatically initiate on either high drywell pressure or low RPV level. The crew will manually initiate Division 1 ECCS using the lock-collared pushbutton **(CT-3)**.

The exercise ends when emergency depressurization is complete and RPV level restoration is being controlled.

Critical Task	**(CT-1) Inhibit ADS prior to automatic ADS valve opening during a LOCA.**	**(CT-2) When RPV level lowers to -160” wide range and cannot be maintained above -191” CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191” CFZ. (Momentary shrink below -191” due to automatic SRV closure does not constitute failure of this critical task.)**
EVENT	5	5
Safety Significance	Permitting automatic ADS initiation may be undesirable for the following reasons: • ADS actuation can impose a severe thermal transient on the RPV and may complicate efforts to control RPV water level. • If only steam-driven systems are available for injection, ADS actuation may directly lead to loss of adequate core cooling and subsequent core damage. • The conditions assumed in the design of the ADS actuation logic (e.g., no operator action for 115 seconds after event initiation) may not exist when the actions specified in this step are being performed. • The operating crew can draw on much more information than is available to the ADS logic (e.g., equipment out of service for maintenance, operating experience with certain systems, probability of restoration of off-site power, etc.) and can better judge, based on instructions contained in the EPGs/SAGs, when and how to depressurize the RPV. Defeating the logic relieves the operating crew of the task of detecting timer initiation during execution of the more complex steps of Contingency #1 and precludes unnecessary and unwanted automatic initiations. Subsequent steps provide explicit and detailed instructions for controlling RPV water level and specify when emergency depressurization is appropriate.	The MSCWL 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. When water level decreases below MSCWL with injection, clad temperatures may exceed 1500°F.
Cueing	ADS Timer initiated alarm on panel P601/19A-A1	Wide range indication (SPDS and PDS) falls to -160” and lowering trend continues, and, before -160” wide range is reached, initial conditions, field reports, and control room indications convey that adequate high pressure injection cannot be restored before level falls below -191” CFZ.
Performance Indicator	Manipulation of ADS A and ADS B MANUAL INHIBIT switches on panel P601 vertical section.	Manipulation of seven of the eight ADS/SRVs on panel P601: B21-F041K B21-F047L B21-F041F B21-F047A B21-F051C B21-F041D B21-F051A B21-F051B
Performance Feedback	Inhibit switches click into INHIBIT position on panel P601 vertical section. White indicating light on ADS A and ADS B MANUAL INHIBIT switches illuminate. Receipt of ADS/SRV A and ADS/SRV B OOSVC alarms on panel P601/18A-H2 and P601/19A-H2.	Crew will observe ADS/SRV light indication go from green to red, reactor pressure lowering on SPDS and panel P601 indications.

Justification for the chosen performance limit	The 115 second ADS timer allows sufficient time for the crew to recognize and override automatic operation of the system. As long as ADS is inhibited before ADS valves open, reactor pressure will not be reduced.	The MSCWL (-191" CFZ) 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. Emergency depressurization is allowed when level goes below TAF (-160" wide range) and should be performed, if in the judgment of the CRS, level cannot be maintained above -191" CFZ. Since it is intended for the scenario supporting this CT to, early in the event, clearly indicate no high pressure injection systems can be made available to reverse the lowering level trend, the crew will have time to communicate and opens at least seven ADS/SRVs before -191" CFZ.
BWR Owners Group Appendix	App. B, step C1-1	App. B, Contingency #1 Step C1-4

Critical Task	**(CT-3) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually initiates ECCS systems for injection prior to RPV pressure lowering below 300 psig.**	
EVENT	6	
Safety Significance	Failure to recognize the auto initiation not occurring, and failure to take manual action per Conduct of Ops will result in unavailability of safety-related equipment necessary to provide adequate core cooling, otherwise resulting in core damage and a large offsite release.	
Cueing	Indication of ECCS systems not initiating with initiation conditions present: • Indication of Drywell pressure ≥ 1.39 psig or RPV level ≤ -150.3" wide range • White light on LPCS/RHR A INIT RESET pushbutton extinguished on panel P601 • Green light on and red light extinguished on LPCS and RHR A pump handswitches on panel P601	
Performance Indicator	Operator manually initiates Division 1 ECCS by rotating the arming collar and depressing the LPCS/RHR A MAN INIT pushbutton on panel P601.	
Performance Feedback	Red light on and green light extinguished on LPCS and RHR A pump handswitches on panel P601. Rising level trend on indications on panel P601, PDS and SPDS. Rising flow rate on LPCS and/or RHR A flow indicators on panel P601, PDS, and SPDS.	
Justification for the chosen performance limit	Attempting to align high pressure ECCS systems must be performed to determine their availability by the time TAF is reached in order to properly implement EP-2 decision steps regarding restoring and maintaining RPV level. Attempting to align low pressure ECCS systems can only be done one RPV pressure falls below the injection valve RPV pressure permissive and will only be effective once RPV pressure falls below the shutoff head of the respective ECCS pump. The reduction in RPV pressure will normally be via Emergency Depressurization, which is a separate critical task bounded by a minimum RPV level.	
BWR Owners Group Appendix	App. B, Contingency 1, step C1-3	

Facility: Grand Gulf Nuclear Station Scenario No.: 4 Op-Test No.: GGNS 12-2017

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

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

1. Withdraw control rods to 10% Bypass Valve position
2. Respond to a HPCS CST Level LO trip unit failure.
3. Respond to Condensate Pump B trip.
4. Respond to a Bus 15AA Feeder Breaker trip with 15BA4 failure to re-energize.
5. Respond to a Startup Level Control controller failure.
6. Respond to an Aux Steam Tunnel steam leak with failure of one steam line to isolate.
7. Respond to a Feedwater Line B line break inside the Drywell with B21-F065B power loss.
8. Respond to a LPCS logic power failure.

Initial Conditions:

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

Inoperable Equipment: None

Turnover:

- Reactor startup is in progress
    - o Step 90 of Control Rod Movement Sequence is complete
    - o SJAE B is in service
- Condensate System is lined up as follows:
    - o Condensate Pumps B and C in service
    - o Condensate Booster Pump A in service
    - o Reactor Feed Pump A in service at approximately 950 psig discharge pressure
    - o CFFF is in service
    - o 4 Deep-Bed Condensate Demineralizers are in service
- Annunciators P680-4A2-C5, CONT ROD WITHDRAWAL BLOCK, and P680-4A1-A7, CRD DRIVE WTR TO RX ΔP HI, are flagged as expected annunciators

Planned activities for this shift are:

- Withdraw control rods until 10% Bypass Valve position on the lagging valve, then continue raising TURB STM PRESSURE DEMAND setpoint to 935 psig per step 45 of IOI 03-1-01-1, Attachment XV

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 60 minutes

Event No.	Malf. No.	Event Type †	Event Description
1	N/A	N (ATC,CRS)	Withdraw control rods to 10% Bypass Valve position.
2	pte22n654c_a	TS (CRS) I (BOP,CRS)	Respond to a HPCS CST Level Lo trip unit failure.
3	fw115a	C (ATC,CRS) A (CREW)	Respond to a Condensate Pump B trip.
4	(or) di_r21m606a r21142u	TS (CRS) C(BOP,CRS) A (CREW)	Respond to a Bus 15AA feeder breaker trip with a failure of 15BA4 to re-energize.
5	fw124	C(ATC,CRS)	Respond to a Startup Level Control controller failure.
6	ms066a ms183a ms184a	M (CREW)	Respond to an Aux Steam Tunnel steam leak with failure of one steam line to isolate. * **(CT-1) When MSIVs fail to isolate, manually scram the reactor and close the MSIVs prior to Steam Tunnel temperature exceeding 250°F (Max Safe Temperature)**
7	b21f065b_i fw171b	C (BOP,CRS)	Respond to a Feedwater Line B line break inside Drywell with a B21-F065B power loss. * **(CT-2) When RPV level lowers to -160” wide range and cannot be maintained above -191” CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191” CFZ. (Momentary shrink below -191” due to automatic SRV closure does not constitute failure of this critical task.)**
8	r21219	C (ATC,CRS)	Respond to a LPCS logic power failure. * **(CT-3) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually initiates ECCS systems for injection prior to RPV pressure lowering below 300 psig**
†	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec		
*	**Critical Task** (As defined in NUREG 1021 Appendix D)		

CREW notation for Abnormal (A) and Major (M) events denotes ATC, BOP, and CRS are credited.

Quantitative Attributes Table

Attribute	E3-304-1 Target	Actual	Description
Malfunctions after EOP entry	1-2	2	• Respond to a Feedwater Line B line break inside the Drywell with a B21-F065B power loss • Respond to a LPCS Logic power failure
Abnormal Events	2-4	2	• Respond to a 15AA Feeder Breaker trip with a failure of 15BA4 to re-energize • Respond to a Condensate Pump B trip
Major Transients	1-2	1	• Respond to an Aux Steam Tunnel steam leak with a failure of one steam line to isolate
EOP entries requiring substantive action	1-2	3	• EP-4 • EP-2 • EP-3
EOP contingencies requiring substantive action	0-2	2	• EP-2 Alternate Level Control • EP-2 Emergency Depressurization
EOP based Critical Tasks	2-3	3	• **(CT-1) When MSIVs fail to isolate, manually scram the reactor and close the MSIVs prior to Steam Tunnel temperature exceeding 250°F (Max Safe Temperature)** • **(CT-2) When RPV level lowers to -160” wide range and cannot be maintained above -191” CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191” CFZ. (Momentary shrink below -191” due to automatic SRV closure does not constitute failure of this critical task.)** • **(CT-3) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually initiates ECCS systems for injection prior to RPV pressure lowering below 300 psig**
Normal Events	N/A	1	• Withdraw control rods to 10% Bypass Valve position
Reactivity Manipulations	N/A	0	• N/A
Instrument / Component failures	N/A	6	• Respond to a HPCS CST Level LO trip unit failure • Respond to Condensate Pump B trip • Respond to a 15AA Feeder Breaker trip with a failure of 15BA4 to re-energize • Respond to a Startup Level Control controller failure • Respond to a Feedwater Line B line break inside the Drywell with a B21-F065B power loss • Respond to a LPCS logic power failure
Total Malfunctions	N/A	7	• Respond to a HPCS CST Level LO trip unit failure • Respond to Condensate Pump B trip • Respond to a 15AA Feeder Breaker trip with a failure of 15BA4 to re-energize • Respond to a Startup Level Control controller failure • Respond to an Aux Steam Tunnel steam leak with failure of one steam line to isolate • Respond to a Feedwater Line B line break inside the Drywell with a B21-F065B power loss • Respond to a LPCS logic power failure

Top 10 systems and operator actions important to risk that are tested:

ADS (Event 7)
RHR (Event 8)
ESF Power (R20) (Event 4)
Condensate (Event 3)
Failure to manually depressurize with ADS/SRVs (Event 7)
Failure to align alternate power to 4.16 KV or 6.9KV buses (Event 4)

SCENARIO ACTIVITIES:

The plant is operating at 4% power.

Event 1

After the crew assumes the shift, the ATC will withdraw control rods in accordance with Control Rod Movement Sequence.

Event 2 **(Triggered automatically)**

When Control Rod 28-53GN is withdrawn to position 08, HPCS CST Level LO trip unit will fail high. CRS will direct BOP to transfer HPCS suction from CST to Suppression Pool. CRS will enter TS 3.3.5.1 Condition A applies and, using Table 3.3.5.1-1, enter TS 3.3.5.1 Condition D.

Event 3 **(Triggered by Lead Examiner)**

After Tech Specs are addressed, Condensate Pump B will trip. CRS will enter 05-1-02-V-7, Feedwater System Malfunctions ONEP, and direct ATC to start Condensate Pump A.

Event 4 **(Triggered by Lead Examiner)**

After actions of Condensate Pump B trip are complete, Bus 15AA Feeder Breaker from ESF Transformer 11, 152-1514, will trip. Division 1 Diesel Generator will automatically start and supply Bus 15AA. BOP will recognize the failure of 15BA4 to re-energize. CRS will enter 05-1-02-III-5, Automatic Isolations ONEP and 05-1-02-III-1, Inadequate Decay Heat Removal ONEP. BOP will restore Instrument Air to CTMT by opening P53-F001. CRS will enter TS 3.8.7 Condition A.

NOTE: CRS is not expected to formulate plans for recovery of Fuel Pool Cooling and Cleanup or Reactor Water Cleanup systems within the time frame of this scenario.

Event 5 **(Triggered by Lead Examiner)**

After Tech Specs are addressed, the Startup Level Control Controller will begin failing low, resulting in RPV level lowering. ATC will take manual control of the Startup Level Controller and restore RPV level to normal band. CRS will enter 05-1-02-V-7, Feedwater System Malfunctions ONEP.

Event 6 **(Triggered by Lead Examiner)**

When RPV level has been returned to normal and stabilized, a steam leak in the Auxiliary Building Steam Tunnel will occur. The 'A' Steam Line will fail to isolate. The CRS will enter EP-4 and direct the ATC to manually scram the reactor and the BOP to manually close B21-F022A, INBD MSIV, and B21-F028A, OTBD MSIV **(CT-1)**. When the reactor is scrammed, the CRS will enter EP-2.

Event 7 **(Triggered automatically)**

Five minutes after the reactor is scrammed, an unisolable Feedwater Line 'B' break in the Drywell will occur. The BOP will secure all Condensate Pumps and close B21-F065B, FW INL SHUTOFF VLV. B21-F065B will not close due to a power loss when its CLOSE handswitch is depressed. ESF Bus 16AB will lockout and HPCS will trip when Drywell pressure rises to 1.39 psig. When CRS determines there are insufficient high pressure injection sources to maintain RPV level above -160" wide, enters Alternate Level Control contingency of EP-2. When RPV level lowers to -160" wide range, the crew will emergency depressurize the RPV using ADS/SRVs **(CT-2)** and restore RPV level with Division 1 ECCS systems.

Event 8 **(Triggered automatically)**

When Drywell pressure rises to 1.39 psig, a LPCS logic power failure will occur. Crew will respond using ARI 04-1-02-1H13-P601-21A-H8, LPCS SYS OOSVC, and manually align Div 1 ECCS systems for injection to the RPV **(CT-3)**.

The exercise ends when emergency depressurization is complete and RPV level restoration is being controlled.

Critical Task	**(CT-1) When MSIVs fail to isolate, manually scram the reactor and close the MSIVs prior to Steam Tunnel temperature exceeding 250°F (Max Safe Temperature).**	**(CT-2) When RPV level lowers to -160" wide range and cannot be maintained above -191" CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191" CFZ. (Momentary shrink below -191" due to automatic SRV closure does not constitute failure of this critical task.)**
EVENT	6	7
Safety Significance	If a primary system is discharging into the secondary containment when this step of the procedure is reached, one of three conditions must exist: • A primary system break cannot be isolated because system operation is required to assure adequate core cooling or to shut down the reactor. • No isolation valves exist upstream of a primary system break, or if isolation valves do exist, they cannot be closed because of some mechanical/ electrical/pneumatic failure. • The source of the discharge cannot be determined. Since the RPV is the only significant source of heat, other than a fire, which might cause area temperatures to increase to their maximum safe operating values, the action of manually scramming the reactor should terminate increasing secondary containment temperatures. If temperatures in any one of the areas listed in Table SC-1 of the Secondary Containment Control guideline approach their maximum safe operating value, adequate core cooling, containment integrity, safety of personnel, or continued operability of equipment required to perform EPG actions can no longer be assured.	The MSCWL 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. When water level decreases below MSCWL with injection, clad temperatures may exceed 1500°F.
Cueing	Main Steam Tunnel temperature rising on PDS. Main Steam Tunnel temperature alarms on panel P601. MSIV open position indication on panel P601 and panel P858.	Wide range indication (SPDS and PDS) falls to -160" and lowering trend continues, and, before -160" wide range is reached, initial conditions, field reports, and control room indications convey that adequate high pressure injection cannot be restored before level falls below -191" CFZ.
Performance Indicator	Operator places the Reactor Mode Switch to SHUTDOWN on panel P680. Operator manipulates switches for MSIVs for Steam Line 'A' to CLOSE on panel P601.	Manipulation of seven of the eight ADS/SRVs on panel P601: B21-F041K B21-F047L B21-F041F B21-F047A B21-F051C B21-F041D B21-F051A B21-F051B
Performance Feedback	RPS Group lights de-energized on panel P680. Control Rod full –in indication on panel P680. Reactor power trend on nuclear instrumentation on panel P680. Green light indication energized and red light indication off for MSIVs for Steam Line 'A' on panel P601 and P858.	Crew will observe ADS/SRV light indication go from green to red, reactor pressure lowering on SPDS and panel P601 indications.

Justification for the chosen performance limit	If temperatures in any one of the areas listed in Table SC-1 of the Secondary Containment Control guideline approach their maximum safe operating value, adequate core cooling, containment integrity, safety of personnel, or continued operability of equipment required to perform EPG actions can no longer be assured.	The MSCWL (-191" CFZ) 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. Emergency depressurization is allowed when level goes below TAF (-160" wide range) and should be performed, if in the judgment of the CRS, level cannot be maintained above -191" CFZ. Since it is intended for the scenario supporting this CT to, early in the event, clearly indicate no high pressure injection systems can be made available to reverse the lowering level trend, the crew will have time to communicate and open 7 of 8 ADS/SRVs before -191" CFZ.
BWR Owners Group Appendix	App. B, step SC/T-4 and SC/T-4.1	App. B, Contingency #1 Step C1-4

Critical Task	**(CT-3) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually aligns ECCS systems for injection prior to RPV pressure lowering below 300 psig.**	
EVENT	8	
Safety Significance	Failure to recognize the auto initiation not occurring, and failure to take manual action per Conduct of Ops will result in unavailability of safety-related equipment necessary to provide adequate core cooling, otherwise resulting in core damage and a large offsite release.	
Cueing	Indication of ECCS systems not initiating with initiation conditions present: • Indication of Drywell pressure ≥ 1.39 psig or RPV level ≤ -150.3" wide range • White light on LPCS/RHR A INIT RESET pushbutton extinguished on panel P601 • Green light on and red light extinguished on LPCS and RHR A pump handswitches on panel P601 • LPCS SYS OOSVC annunciator on panel P601	
Performance Indicator	Operator manually manipulates switches for Div 1 ECCS pumps and directs operators to manually open Div 1 ECCS injection valves from Division 1 Remote Shutdown Panel (RHR A) and locally (LPCS).	
Performance Feedback	Red light on and green light extinguished on LPCS and/or RHR A pump and valve handswitches on panel P601. Rising level trend on indications on panel P601, PDS and SPDS. Rising flow rate on LPCS and/or RHR A flow indicators on panel P601, PDS, and SPDS	
Justification for the chosen performance limit	Attempting to align high pressure ECCS systems must be performed to determine their availability by the time TAF is reached in order to properly implement EP-2 decision steps regarding restoring and maintaining RPV level. Attempting to align low pressure ECCS systems can only be done once RPV pressure falls below the injection valve RPV pressure permissive and will only be effective once RPV pressure falls below the shutoff head of the respective ECCS pump. The reduction in RPV pressure will normally be via Emergency Depressurization, which is a separate critical task bounded by a minimum RPV level.	
BWR Owners Group Appendix	App. B, Contingency 1, step C1-3	