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July 23, 2015

Kelly Clayton, Chief Examiner
U.S. Nuclear Regulatory Commission, Region IV
1600 East Lamar Blvd
Arlington, TX 76011-4511

SUBJECT: NRC INITIAL EXAMINATION OUTLINE

GEXO: 2015/00038

Dear Mr. Clayton,

Enclosed is the NRC Initial Examination Outline for the Initial License Examination to be administered the week of December 7, 2015 (written exam is scheduled for December 4, 2015). This class consists of 7 RO candidates, 1 SRO Instant candidate, and 2 SRO Upgrade candidates.

The following NUREG 1021 Forms are enclosed with the supporting documentation:

ES-201-2; 301-1; 301-2; 301-3, 301-4; 301-5; D-1 for 3 scenarios
ES-401-1, 401-3, 401-4

Also enclosed are the current Exam Security Agreement (ES-201-3) and schedule for the Operating Test.

Please contact Gabe or Steve at (601) 437-2255 if you have any questions or need any further materials.

Sincerely,

Elizabeth Meaders
Manager, Training & Development
Grand Gulf Nuclear Station

Chad Rogers
Superintendent, FIN Team
Facility Representative
Grand Gulf Nuclear Station

Facility: Grand Gulf Nuclear Station Date of Exam: 12/4/2015

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

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

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

G* Generic K/As

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4									
295003 Partial or Complete Loss of AC / 6			X				**Knowledge of the reasons for the following responses as they apply to a partial or complete loss of A.C. power:** AK3.01: Manual and auto bus transfer CFR: 41.5	3.3	1
295004 Partial or Total Loss of DC Pwr / 6					X		**Ability to determine and/or interpret the following as they apply to a partial or complete loss of D.C. power:** AA2.03: Battery voltage CFR: 41.10	2.8	2
295005 Main Turbine Generator Trip / 3				X			**Ability to operate and/or monitor the following as they apply to main turbine generator trip:** AA1.02: RPS CFR: 41.7	3.6	3
295006 SCRAM / 1		X					**Knowledge of the interrelations between SCRAM and the following:** AK2.06: Reactor power CFR: 41.7	4.2*	4
295016 Control Room Abandonment / 7						X	**For control room abandonment:** G2.4.35: Knowledge of local auxiliary operator tasks during an emergency and the resultant operation effects. CFR: 41.10	3.8	5
295018 Partial or Total Loss of CCW / 8					X		**Ability to determine and/or interpret the following as they apply to partial or complete loss of component cooling water:** AA2.01: Component temperatures CFR: 41.10	3.3	6
295019 Partial or Total Loss of Inst. Air / 8			X				**Knowledge of the reasons for the following responses as they apply to partial or complete loss of instrument air:** AK3.02: Standby air compressor operations CFR: 41.5	3.5	7
295021 Loss of Shutdown Cooling / 4		X					**Knowledge of the interrelations between loss of shutdown cooling and the following:** AK2.01: Reactor water temperature CFR: 41.7	3.6	8

295023 Refueling Acc / 8	X						**Knowledge of the operational implications of the following concepts as they apply to refueling accidents:** AK1.03: Inadvertent criticality CFR: 41.8-41.10	3.7	9
295024 High Drywell Pressure / 5					X		**Ability to determine and/or interpret the following as they apply to high drywell pressure:** EA2.02: Drywell temperature CFR: 41.10	3.9	10
295025 High Reactor Pressure / 3		X					**Knowledge of the interrelations between high reactor pressure and the following:** EK2.09: Reactor power CFR: 41.7	3.9	11
295026 Suppression Pool High Water Temp. / 5				X			**Ability to operate and/or monitor the following as they apply to suppression pool high water temperature:** EA1.03: Temperature monitoring CFR: 41.7	3.9*	12
295027 High Containment Temperature / 5	X						**Knowledge of the operational implications of the following concepts as they apply to high containment temperature (Mark III containment only):** EK1.02: Reactor water level measurement: Mark-III CFR: 41.8-41.10	3.0	13
295028 High Drywell Temperature / 5						X	**For high drywell temperature:** G2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. CFR: 41.5	4.4	14
295030 Low Suppression Pool Wtr Lvl / 5				X			**Ability to opeatoe and/or monitor the following as they apply to low suppression pool water level:** EA1.06: Condensate storage and transfer (make-up to the suppression pool): Plant-specific CFR: 41.7	3.4	15
295031 Reactor Low Water Level / 2			X				**Knowledge of the reasons for the following responses as they apply to reactor low water level:** EK3.02: Core coverage CFR: 41.5	4.4*	16
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1	X						**Knowledge of the operational implications of the following concepts as they apply to SCRAM condition present and reactor power above APRM downscale or unknown:** EK1.07: Shutdown margin CFR: 41.08-41.10	3.4	17

295038 High Off-site Release Rate / 9		X					**Knowledge of the interrelations between high off-site release rate and the following:** EK2.03: Plant ventilation systems CFR: 41.7	3.6	18
600000 Plant Fire On Site / 8					X		**Ability to determine and interpret the following as they apply to plant fire on site:** AA2.17: Systems that may be affected by the fire	3.1	19
700000 Generator Voltage and Electric Grid Disturbances / 6						X	**For generator voltage and electric grid disturbances:** G2.4.45: Ability to prioritize and interpret the significance of each annunciator or alarm CFR: 41.10	4.1	20
K/A Category Totals:	3	4	3	3	4	3	Group Point Total:		20/7

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / 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									
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2	X						**Knowledge of the operational implications of the following concepts as they apply to low reactor water level:** AK1.05: Natural circulation CFR: 41.8-41.10	3.3	21
295010 High Drywell Pressure / 5									
295011 High Containment Temp / 5						X	**For High Containment Temperature:** G2.4.1 Knowledge of EOP entry conditions and immediate action steps. CFR: 41.10	4.6	22
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5									
295014 Inadvertent Reactivity Addition / 1				X			**Ability to operate and/or monitor the following as they apply to inadvertent reactivity addition:** AA1.05: Neutron monitoring system CFR: 41.7	3.9	23
295015 Incomplete SCRAM / 1			X				**Knowledge of the reasons for the following responses as they apply to incomplete SCRAM:** AK3.01: Bypassing rod insertion blocks CFR: 41.5	3.4	24
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		X					**Knowledge of the interrelations between secondary containment ventilation high radiation and the following:** EK2.04: Secondary containment ventilation CFR: 41.7	3.9	25

295035 Secondary Containment High Differential Pressure / 5					X		**Ability to determine and/or interpret the following as they apply to secondary containment high differential pressure:** EA2.01: Secondary containment pressure: Plant-Specific CFR: 41.8-41.10	3.8	26
295036 Secondary Containment High Sump/Area Water Level / 5		X					**Knowledge of the interrelations between secondary containment high sump area water level and the following:** EK2.03: Radwaste CFR: 41.7	2.8	27
500000 High CTMT Hydrogen Conc. / 5									
K/A Category Point Totals:	1	2	1	1	1	1	Group Point Total:		7/3

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

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	#
203000 RHR/LPCI: Injection Mode							X					**Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: Injection Mode (Plant Specific) controls including:** A1.01: Reactor water level CFR: 41.5	4.2*	28
205000 Shutdown Cooling				X								**Knowledge of shutdown cooling system (RHR shutdown cooling mode) design feature(s) and/or interlocks which provide for the following:** K4.03: Low reactor water level: Plant-Specific CFR: 41.7	3.8	29
~~206000 HPCI~~														
~~207000 Isolation (Emergency) Condenser~~														
209001 LPCS	X											**Knowledge of the physical connections and/or cause-effect relationships between low pressure core spray system and the following:** K1.02: Torus/suppression pool CFR: 41.2-41.9	3.4	30
209002 HPCS									X			**Ability to monitor automatic operations of the high pressure core spray system (HPCS) including:** A3.03: System pressure: BWR-5,6 CFR: 41.7	3.6	31
211000 SLC						X						**Knowledge of the effect that a loss or malfunction of the following will have on the standby liquid control system:** K6.03: A.C. power CFR: 41.7	3.2	32
212000 RPS					X							**Knowledge of the operational implications of the following concepts as they apply to reactor protection system:** K5.02: Specific logic arrangements CFR: 41.5	3.3	33

215003 IRM											X	**For IRM system:** G2.1.32: Ability to explain and apply system limits and precautions. CFR: 41.10	3.8	34
215004 Source Range Monitor	X								X			**Knowledge of the physical connections and/or cause-effect relationships between source range monitor (SRM) system and the following:** K1.02: Reactor manual control CFR: 41.2-41.9 **Ability to monitor automatic operations of the source range monitor system including:** A3.04: Control rod block status CFR: 41.7	3.4 3.6	35 36
215005 APRM / LPRM							X					**Ability to predict and/or monitor changes in parameters associated with operating the average power range monitor/local power range monitor system:** A1.06: Recirculation flow control valve position: Plant-Specific CFR: 41.5	3.1	37
217000 RCIC			X		X							**Knowledge of the effect that a loss or malfunction of the reactor core isolation cooling system (RCIC) will have on the following:** K3.03: Decay heat removal CFR: 41.7 **Knowledge of the operational implications of the following concepts as they apply to reactor core isolation cooling system (RCIC):** K5.02: Flow indication CFR: 41.5	3.5 3.1	38 39
218000 ADS		X										**Knowledge of electrical power supplies to the following:** K2.01 ADS logic CFR: 41.7	3.1*	40
223002 PCIS/Nuclear Steam Supply Shutoff										X		**Ability to manually operate and/or monitor in the control room:** A4.01: Valve closures CFR: 41.7	3.6	41

239002 SRVs						X		X				**Knowledge of the effect that a loss or malfunction of the following will have on the relief/safety valves:** K6.02: Air (Nitrogen) supply: Plant-Specific CFR: 41.7 **Ability to (a) predict the impacts of the following on the relief/safety valves; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.04: ADS actuation CFR: 41.5	3.4 4.1*	42 43
259002 Reactor Water Level Control							X			X		**Ability to predict and/or monitor changes in parameters associated with operating the reactor water level control system controls including:** A1.02: Reactor feedwater flow CFR: 41.5 **Ability to manually operate and/or monitor in the control room:** A4.11: High level lockout reset controls: Plant-Specific CFR: 41.7	3.6 3.5	44 45
261000 SGTS			X									**Knowledge of the effect that a loss or malfunction of the standby gas treatment system will have on the following:** K3.01: Secondary containment and environment differential pressure CFR: 41.7	3.3	46
262001 AC Electrical Distribution											X	**For AC electrical distribution system:** G2.1.19: Ability to use plant computers to evaluate system or component status CFR: 41.10	3.9	47
262002 UPS (AC/DC)								X				**Ability to (a) predict the impacts of the following on the uninterruptable power supply (AC/DC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.01: Under voltage CFR: 41.5	2.6	48
263000 DC Electrical Distribution		X										**Knowledge of electrical power supplies to the following:** K2.01: Major D.C. loads CFR: 41.7	3.1	49

264000 EDGs									X			**Ability to monitor automatic operations of the emergency generators (diesel/jet) including:** A3.06: Cooling water system operation CFR: 41.7	3.1	50
300000 Instrument Air				X								**Knowledge of instrument air system design feature(s) and or interlocks which provide for the following:** K4.02: Cross-over to other air systems CFR: 41.7	3.0	51
400000 Component Cooling Water	X										X	**Knowledge of the physical connections and/or cause-effect relationships between CCWS and the following:** K1.02: Loads cooled by CCWS CFR: 41.2-41.9 **For the component cooling water system:** G2.2.22: Knowledge of limiting conditions for operations and safety limits CFR: 41.5	3.2 4.0	52 53
K/A Category Point Totals:	3	2	2	2	2	2	3	2	3	2	3	Group Point Total:		26/5

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

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			X									**Knowledge of the effect that a loss or malfunction of the control rod drive hydraulic system will have on the following:** K3.02: Reactor water level CFR: 41.7	2.6	54
201002 RMCS														
201003 Control Rod and Drive Mechanism									X			**Ability to monitor automatic operations of the control rod and drive mechanism including:** A3.01: Control rod position CFR: 41.7	3.7	55
201004 RSCS														
201005 RCIS								X				**Ability to (a) predict the impacts of the following on the rod control and information system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.12: Rod uncoupled: BWR-6 CFR: 41.5	3.7	56
201006 RWM														
202001 Recirculation							X					**Ability to predict and/or monitor changes in parameters associated with operating the recirculation system controls including:** A1.05 Reactor power CFR: 41.5	3.9	57
202002 Recirculation Flow Control														
204000 RWCU											X	**For the reactor water cleanup system:** G2.1.28: Knowledge of the purpose and function of major system components and controls. CFR: 41.7	4.1	58
214000 RPIS														
215001 Traversing In-core Probe														
215002 RBM														
216000 Nuclear Boiler Inst.														
219000 RHR/LPCI: Torus/Pool Cooling Mode														

223001 Primary CTMT and Aux.					X							**Knowledge of the operational implications of the following concepts as they apply to primary containment system and auxiliaries:** K5.08: Pressure measurement CFR: 41.5	2.7	59
226001 RHR/LPCI: CTMT Spray Mode				X								**Knowledge of RHR/LPCI: containment spray system mode design feature(s) and/or interlocks which provide for the following:** K4.11: Prevention of leakage to the environment through system heat exchanger CFR: 41.7	2.7	60
230000 RHR/LPCI: Torus/Pool Spray Mode														
233000 Fuel Pool Cooling/Cleanup	X											**Knowledge of the physical connections and/or cause-effect relationships between fuel pool cooling and clean-up and the following:** K1.02: Residual heat removal system: Plant-Specific CFR: 41.2-41.9	2.9	61
234000 Fuel Handling Equipment														
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator										X		**Ability to manually operate and/or monitor in the control room:** A4.07: Main stop/throttle valves (operation) CFR: 41.7	3.5	62
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:** K6.02: Reactor/turbine pressure control system: Plant-Specific CFR: 41.7	3.5	63
256000 Reactor Condensate														
259001 Reactor Feedwater														
268000 Radwaste														
271000 Offgas			X									**Knowledge of the effect that a loss or malfunction of the offgas systems will have on the following:** K3.01: Condenser vacuum CFR: 41.5	3.5	64
272000 Radiation Monitoring														

286000 Fire Protection		X										**Knowledge of electrical power supplies to the following:** K2.02: Pumps CFR: 41.7	2.9*	65
288000 Plant Ventilation														
290001 Secondary CTMT														
290003 Control Room HVAC														
290002 Reactor Vessel Internals														
K/A Category Point Totals:	1	1	2	1	1	1	1	1	1	1	1	Group Point Total:		12/3

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1	295024 EA 2.08	De-selected due to a lack of adequate distracters.
1/2	295036 EK 2.02	De-selected, N/A for GGNS.
2/1	212000 K 5.01	De-selected due to inability to prepare a psychometrically sound question related to the K/A.
2/1	217000 K 5.03	De-selected, N/A for GGNS.
2/1	400000 K 1.04	De-selected due to a lack of adequate distracters.
2/2	233000 K 1.01	De-selected, N/A for GGNS.
2/2	286000 K 2.03	De-selected due to Low Operational value for discriminatory RO level question.

SYSTEMS DELETED

201002 Reactor Manual Control System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

201004 Rod Sequence Control System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

201006 Rod Worth Minimizer System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

214000 Rod Position Information System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

215002 Rod Block Monitor System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

206000 High Pressure Core Injection (HPCI) - This system is not incorporated into the BWR 6 design.

207000 Isolation (Emergency) Condenser - This system is not incorporated into the BWR 6 design. This was replaced by the Mark III Containment Suppression Pool.

230000 RHR/LPCI: Torus/Pool Spray Mode - This system is not incorporated into the BWR 6 Mark III Containment design.

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

E/APE # / Name / Safety Function	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						X	**For partial or complete loss of forced core flow circulation:** G2.4.8: Knowledge of how abnormal operating procedures are used in conjunction with EOPs. CFR: 43.5	4.5	76
295003 Partial or Complete Loss of AC / 6									
295004 Partial or Total Loss of DC Pwr / 6									
295005 Main Turbine Generator Trip / 3									
295006 SCRAM / 1									
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. CFR: 43.5	4.3	77
295018 Partial or Total Loss of CCW / 8									
295019 Partial or Total Loss of Inst. Air / 8						X	**For a partial or total loss of instrument air:** G2.4.11: Knowledge of abnormal condition procedures. CFR: 43.5	4.2	78
295021 Loss of Shutdown Cooling / 4						X	**For loss of shutdown cooling:** G2.4.47: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. CFR: 43.5	4.2	79
295023 Refueling Acc / 8									
295024 High Drywell Pressure / 5									
295025 High Reactor Pressure / 3									
295026 Suppression Pool High Water Temp. / 5									
295027 High Containment Temperature / 5									
295028 High Drywell Temperature / 5									
295030 Low Suppression Pool Wtr Lvl / 5						X	**For low suppression pool water level:** G2.4.18: Knowledge of the specific bases for EOPs. CFR: 43.1	4.0	80
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					X		**Ability to determine and/or interpret the following as they apply to high off-site release rate:** AA2.04: Source of off-site release CFR: 43.5	4.3*	81
600000 Plant Fire On Site / 8									
700000 Generator Voltage and Electric Grid Disturbances / 6					X		**Ability to determine and/or interpret the following as they apply to generator voltage and electric grid disturbances:** AA2.06: Generator frequency limitations CFR: 43.5	3.5	82
K/A Category Totals:					3	4	Group Point Total:		20/7

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / 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									
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2									
295010 High Drywell Pressure / 5					X		**Ability to determine and/or interpret the following as they apply to high drywell pressure:** AA2.06: Drywell temperature CFR: 43.5	3.6	83
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5									
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1									
295017 High Off-site Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1					X		**Ability to determine and/or interpret the following as they apply to loss of CRD pumps:** AA2.03: CRD mechanism temperatures CFR: 43.5	3.2	84
295029 High Suppression Pool Wtr Lvl / 5						X	**For high suppression pool water level:** G2.4.6: Knowledge of EOP mitigation strategies CFR: 43.5	4.7	85
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									

K/A Category Point Totals:					2	1	Group Point Total:		7/3

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

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	#
203000 RHR/LPCI: Injection Mode														
205000 Shutdown Cooling														
~~206000 HPCI~~														
~~207000 Isolation (Emergency) Condenser~~														
209001 LPCS								X				**Ability to (a) predict the impacts of the following on the low pressure core spray system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.02: Valve closures CFR: 41.5	3.2	86
209002 HPCS														
211000 SLC														
212000 RPS											X	**For RPS:** G2.2.22: Knowledge of limiting conditions for operations and safety limits CFR: 43.2	4.7	87
215003 IRM														
215004 Source Range Monitor														
215005 APRM / LPRM														
217000 RCIC														
218000 ADS														
223002 PCIS/Nuclear Steam Supply Shutoff														
239002 SRVs														
259002 Reactor Water Level Control								X				**Ability to (a) predict the impacts of the following on the reactor water level control system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.04: RFP runout condition: Plant-Specific CFR: 41.5	3.1	88
261000 SGTS														

262001 AC Electrical Distribution														
262002 UPS (AC/DC)											X	**For the UPS:** G2.4.11: Knowledge of abnormal condition procedures. CFR: 43.5	4.2	89
263000 DC Electrical Distribution														
264000 EDGs														
300000 Instrument Air														
400000 Component Cooling Water								X				**Ability to (a) predict the impacts of the following on the component cooling water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.01: Loss of CCW pump CFR: 41.5	3.4	90
K/A Category Point Totals:								3			2	Group Point Total:		26/5

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	IR	#
201001 CRD Hydraulic														
201002 RMCS														
201003 Control Rod and Drive Mechanism														
~~201004 RSCS~~														
201005 RCIS														
201006 RWM														
202001 Recirculation														
202002 Recirculation Flow Control								X				**Ability to (a) predict the impacts of the following on the recirculation flow control system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.09: Recirculation flow mismatch. Plan Specific CFR: 41.5	3.1	91
204000 RWCU														
214000 RPIS														
~~215001 Traversing In-core Probe~~														
215002 RBM														
216000 Nuclear Boiler Inst.														
219000 RHR/LPCI: Torus/Pool Cooling Mode								X				**Ability to (a) predict the impacts of the following on the RHR/LPCI: Torus/suppression pool cooling mode; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.09: Inadequate room cooling CFR: 41.5	2.9	92
223001 Primary CTMT and Aux.														
226001 RHR/LPCI: CTMT Spray Mode														
230000 RHR/LPCI: Torus/Pool Spray Mode														
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment														
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator														
245000 Main Turbine Gen. / Aux.														

256000 Reactor Condensate														
259001 Reactor Feedwater											X	**For Reactor Feedwater:** G2.2.38: Knowledge of conditions and limitations in the facility license. CFR: 43.1	4.5	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														
K/A Category Point Totals:								2			1	Group Point Total:		12/3

Tier / Group	Randomly Selected K/A	Reason for Rejection
1/1	295016 AA 2.07	De-selected, N/A for GGNS.
1/2	295010 AA 2.04	De-selected, N/A for GGNS.
2/1	262002 2.4.20	De-selected due to inability to write discriminatory SRO level question for this K/A.
2/2	202002 A 2.04	De-selected, N/A for GGNS.

SYSTEMS DELETED

201002 Reactor Manual Control System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

201004 Rod Sequence Control System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

201006 Rod Worth Minimizer System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

214000 Rod Position Information System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

215002 Rod Block Monitor System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

206000 High Pressure Core Injection (HPCI) - This system is not incorporated into the BWR 6 design.

207000 Isolation (Emergency) Condenser - This system is not incorporated into the BWR 6 design. This was replaced by the Mark III Containment Suppression Pool.

230000 RHR/LPCI: Torus/Pool Spray Mode - This system is not incorporated into the BWR 6 Mark III Containment design.

Facility: Grand Gulf Nuclear Station		Date of Exam: 12/4/2015				
Category	K/A #	Topic	RO		SRO-Only	
			IR	#	IR	#
1. Conduct of Operations	2.1.3	Knowledge of shift or short-term relief turnover practices. CFR: 41.10	3.7	66		
	2.1.13	Knowledge of facility requirements for controlling vital/controlled access. CFR: 41.10	2.5	67		
	2.1.14	Knowledge of criteria or conditions that require plant-wide announcements, such as pump starts, reactor trips, mode changes, etc. CFR: 41.10	3.1	68		
	2.1.4	Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc. CFR: 43.2			3.8	94
	2.1.37	Knowledge of procedures, guidelines, or limitations associated with reactivity management. CFR: 43.6			4.6	95
	Subtotal		3		2	
2. Equipment Control	2.2.7	Knowledge of the process for conducting special or infrequent tests. CFR: 41.10	2.9	69		
	2.2.14	Knowledge of the process for controlling equipment configuration or status. CFR: 41.10	3.9	70		
	2.2.35	Ability to determine Technical Specification Mode of Operation. CFR: 41.7	3.6	71		
	2.2.6	Knowledge of the process for making design or operating changes to the facility. CFR: 43.3			3.6	96
	2.2.11	Knowledge of the process for controlling temporary design changes. CFR: 43.3			3.3	97
	Subtotal		3		2	

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. CFR: 41.12	3.4	72		
	2.3.15	Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. CFR: 41.12	2.9	73		
	2.3.11	Ability to control radiation releases. CFR: 43.4			4.3	98
	2.3.					
	Subtotal		2		1	
4. Emergency Procedures / Plan	2.4.12	Knowledge of general operating crew responsibilities during emergency operations. CFR: 41.10	4.0	74		
	2.4.39	Knowledge of RO responsibilities in emergency plan implementation. CFR: 41.10	3.9	75		
	2.4.37	Knowledge of the lines of authority during implementation of the emergency plan. CFR: 41.10			4.1	99
	2.4.42	Knowledge of emergency response facilities. CFR: 41.10			3.8	100
	Subtotal					
Tier 3 Point Total			2	10	2	7

Facility: **Grand Gulf Nuclear Station** Date of Examination: 12/7/2015
Examination Level: RO ☒ SRO ☐ Operating Test Number: LOT-2015

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R - M	Loss of Shutdown Cooling, Time to 200F Determination GJPM-OPS-2015-AR1
Conduct of Operations	R - D	Fire Door Surveillance GJPM-OPS-2015-AR2
Equipment Control	R - N	Determine Affect on SSW for Breaker 72-11A33 GJPM-OPS-2015-AR3
Radiation Control		
Emergency Plan	R - M	Station Blackout Electrical Power Determination GJPM-OPS-2015-AR4

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.

* 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: 12/7/2015
Examination Level: RO ☐ SRO ☒ Operating Test Number: LOT-2015

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R - D	Determine Firewatch Requirements GJPM-OPS-2015-AS1
Conduct of Operations	R - N	Manual On-Line Risk Assessment GJPM-OPS-2015-AS2
Equipment Control	R – D	Tagout Removal Approval GJPM-OPS-2015-AS3
Radiation Control	R - D	Review Liquid Radwaste Discharge Permit GJPM-OPS-2015-AS4
Emergency Plan	R - N	Emergency Classification GJPM-OPS-2015-AS5

NOTE: All items (5 total) are required for SROs. RO applicants require only 4 items unless they are retaking only the administrative topics, when all 5 are required.

* 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: 12/7/2015
Exam Level: RO ☒ SRO-I ☐ SRO-U ☐	Operating Test No.: LOT 2015

Control Room Systems* 8 for RO; 7 for SRO-I; 2 or 3 for SRO-U

	System / JPM Title	Type Code*	Safety Function
a.	202001 A4.02 (3.5/3.4) Reset Recirc FCV Runback GJPM-OPS-2015CR1	A - D - S	1
b.	259001 A3.10 (3.4/3.4), Defeat Feed Pump Level 9 Trips GJPM-OPS-2015CR2	D - C - L	2
c.	239001 A2.11 (4.1/4.3), Slow Closing MSIV GJPM-OPS-2015CR3	A - N - S	3
d.	209002 A1.01 (3.6/3.7), Performing HPCS Quarterly Functional Test, GJPM-OPS-CR4	A - D – EN - S	4
e.	223001 A4.06 (4.0/4.0), Containment Venting GJPM-OPS-2015CR5	C - D - E - L	5
f.	212000 A4.02 (3.6/3.7), Reactor Manual Scram Switch Test, GJPM-OPS-2015CR7	A - D - S	7
g.	400000 A4.01 (3.1/3.0), Manual Start of SSW ‘A’ System, GJPM-OPS-2015CR8	A - M - S	8
h.	272000 A4.02 (3.0/3.0), Perform Area Radiation Monitor Functional Test, GJPM-OPS-2015CR9	D - S	9

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

	System / JPM Title	Type Code*	Safety Function
i.	262001 A2.11 (3.2/3.6), Reset Undervoltage Lockouts on BOP Buses, GJPM-OPS-2015PS1	D - R - L	6
j.	217000 A1.01 (3.7/3.7), Perform RCIC Operation per Shutdown from the Remote Shutdown Panel, GJPM-OPS-2015PS2	N - E – EN - L	4
k.	212000 A2.02 (3.7/3.9), Energize RPS Alternate Feed, GJPM-OPS-2015PS3	D	7

* All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all five SRO-U systems must serve different safety functions; 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 / ≥1 / ≥1 (control room system)
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: **GRAND GULF NUCLEAR STATION**	Date of Examination: 12/7/2015____
Exam Level: RO ☐ SRO-I ☒ SRO-U ☐	Operating Test No.: LOT 2015_____

Control Room Systems* 8 for RO; 7 for SRO-I; 2 or 3 for SRO-U

	System / JPM Title	Type Code*	Safety Function
a.	202001 A4.02 (3.5/3.4) Reset Recirc FCV Runback GJPM-OPS-2015CR1	A - D - S	1
b.	259001 A3.10 (3.4/3.4), Defeat Feed Pump Level 9 Trips GJPM-OPS-2015CR2	D - C - L	2
c.	239001 A2.11 (4.1/4.3), Slow Closing MSIV GJPM-OPS-2015CR3	A - N - S	3
d.	209002 A1.01 (3.6/3.7), Performing HPCS Quarterly Functional Test, GJPM-OPS-CR4	A - D – EN - S	4
e.	223001 A4.06 (4.0/4.0), Containment Venting GJPM-OPS-2015CR5	C - D - E - L	5
f.	212000 A4.02 (3.6/3.7), Reactor Manual Scram Switch Test, GJPM-OPS-2015CR7	A - D - S	7
g.	400000 A4.01 (3.1/3.0), Manual Start of SSW ‘A’ System, GJPM-OPS-2015CR8	A - M - S	8
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)			
h.	262001 A2.11 (3.2/3.6), Reset Undervoltage Lockouts on BOP Buses, GJPM-OPS-2015PS1	D - R - L	6
i.	217000 A1.01 (3.7/3.7), Perform RCIC Operation per Shutdown from the Remote Shutdown Panel, GJPM-OPS-2015PS2	N - E – EN - L	4
j.	212000 A2.02 (3.7/3.9), Energize RPS Alternate Feed, GJPM-OPS-2015PS3	D	7

* All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all five SRO-U systems must serve different safety functions; 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 / ≥1 / ≥1 (control room system)
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: **GRAND GULF NUCLEAR STATION**	Date of Examination: 12/7/2015
Exam Level: RO ☐ SRO-I ☐ SRO-U ☒	Operating Test No.: LOT-2015

Control Room Systems* 8 for RO; 7 for SRO-I; 2 or 3 for SRO-U			
	System / JPM Title	Type Code*	Safety Function
a.	202001 A4.02 (3.5/3.4) Reset Recirc FCV Runback GJPM-OPS-2015CR1	A - D - S	1
b.	239001 A2.11 (4.1/4.3), Slow Closing MSIV GJPM-OPS-2015CR3	A - N - S	3
c.	212000 A4.02 (3.6/3.7), Reactor Manual Scram Switch Test, GJPM-OPS-2015CR7	A - D - S	7
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)			
d.	262001 A2.11 (3.2/3.6), Reset Undervoltage Lockouts on BOP Buses, GJPM-OPS-2015PS1	D - R - L	6
e.	217000 A1.01 (3.7/3.7), Perform RCIC Operation per Shutdown from the Remote Shutdown Panel, GJPM-OPS-2015PS2	N - E – EN - L	4

* All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all five SRO-U systems must serve different safety functions; 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 / ≥1 / ≥1 (control room system)
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

Facility: GGNS																	
Date of Exam: 12/07/15																	
Operating Test No.: LOT-2015																	

APPLICANT	EVENT TYPE	Scenarios												TOTAL	MINIMUM(*)		
		1			2			3									
		CREW POSITION			CREW POSITION			CREW POSITION									
		SRO	ATC	BOP	SRO	ATC	BOP	SRO	ATC	BOP					R	I	U
U1 & U2	RX	0			0			0							1	1	0
	NOR	1			2			1							1	1	1
	I/C	5			6			5							4	4	2
	MAJ	2			1			1							2	2	1
	TS	2			3			2							0	2	2
I1	RX	0				2		0	1						1	1	0
	NOR	1				0		1	0						1	1	1
	I/C	5				3		5	3						4	4	2
	MAJ	2				1		1	1						2	2	1
	TS	2						2							0	2	2
R1 R3 R5 R7	RX		1				0		1	0					1	1	0
	NOR		0				2		0	1					1	1	1
	I/C		3				3		3	3					4	4	2
	MAJ		2				1		1	1					2	2	1
	TS														0	2	2
R2 R4 R6	RX			0		2			1	0					1	1	0
	NOR			2		0			0	1					1	1	1
	I/C			3		3			3	3					4	4	2
	MAJ			2		1			1	1					2	2	1
	TS														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 instrument or component malfunctions on a 1-for-1 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 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.: NRC LOT 2015

Examiners: ____________ Operators: ____________

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

1. Raise reactor power using Recirc Flow Control.
2. Rotate Power to bus 17AC from ESF 12 to ESF 21.
3. Respond to a loss 17AC with a Diesel Generator failure.
4. Respond to inadvertent RCIC initiation.
5. Respond to Feedwater master level controller failure.
6. Respond to Main Turbine Trip
7. Take actions for a low power ATWS.
8. Respond to a SLC system failure.
9. Respond to a Feedwater line B rupture inside Drywell with inability to isolate

Initial Conditions: Operating at 85% power.

Inoperable Equipment: None

Turnover:

The plant is at 85% following a sequence exchange. SSW A is operating in preparation for weekly chemical addition. Planned activities for this shift are:

- Raise power to 90% (1300 Mwe) per IOI. Allow Reactor Engineering to perform preconditioning.
- Rotate Power to 17AC from ESF 12 to ESF 21..

There is no out of service equipment and EOOS is GREEN. It is a division 2 work week.

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 60 minutes

Event No.	Malf. No.	Event Type †	Event Description
1		N (BOP) R (ATC)	Raise Reactor power to 90% using FCV's (IOI 03-1-01-2 Attachment VIII)
2		N (BOP)	Rotate Power to 17AC from ESF 12 to ESF 21.(SOI 04-1-01-R21-17)
3	DI_1E22M716 n41140c	C (BOP) A (CREW) TS (CRS)	Respond to a loss of 17AC with a failure of Division 3 Diesel Generator (Loss of AC Power ONEP, 05-1-02-I-4; Tech Spec 3.8.1)
4	e51188	C (BOP) R (ATC) C (ATC) TS (CRS) A (CREW)	Respond to inadvertent RCIC initiation. (RCIC SOI 04-1-01-E51-1; Loss of Feedwater Heating ONEP, 05-1-02-V-5, Tech Spec 3.5.3)
5	fw127	C (ATC) A (CREW)	Respond to Feedwater master level controller failure. (Feedwater System Malfunctions ONEP, 05-1-02-V-7)
6	tc093	M/A (CREW)	Respond to Main Turbine Trip, (Main Turbine Trip and Reactor Scram ONEP, EP-2)
7	c11164	M (CREW)	Low power ATWS (EP-2A)
8	c41263	C (CREW)	Respond to a SLC system failure
9	fw171b b21f065b_i	C (CREW)	Feedwater line B rupture inside Drywell with inability to isolate
†	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec		

Quantitative Attributes Table

Normal Events	2	Abnormal Events	4
Reactivity Manipulations	2	Total Malfunctions	7
Instrument/Component Failures	5	EP Entries (Requiring substantive action)	1
Major Transients	2	EP Contingencies	3
Tech Spec Calls	2	Critical Tasks	5

SCENARIO ACTIVITIES:

Raise reactor power using FCVs (Normal/Reactivity):

A. If necessary, call the control room as the Shift Manager and direct the CRS to raise reactor power to 90% using IOI. All notifications have been made.

   1. The crew will raise power using 03-1-01-2, Power Operations IOI, Attachment II, Power Ascension from 60% to Full Power. **(Event 1)**

Rotate power on bus 17AC from transformer ESF 12 to ESF 21

A. After reactor power has been raised to 90% contact CRS to perform power swap for Bus 17AC. **(Event 2)**

Loss of Bus 17AC with a Division 3 Diesel Generator failure:

A. 30 seconds after power has been swapped on bus 17AC, incoming feeder 152-1705 will trip causing a loss of 17AC. Div 3 D/G will fail to start. **(Event 3)**

B. The crew will take the actions per Loss of AC Power ONEP by energizing the 17AC bus from an alternate feeder.

C. If sent to investigate 152-1705, as Electrical Maintenance wait 3 minutes then report problem is unknown and a work order is required.

D. The CRS will determine that TS 3.8.1 Condition B applies and require surveillance 06-OP-1R21-W-0001 Attachment II.

Inadvertent initiation of RCIC:

A. When the crew has addressed all required steps of the Loss of AC Power ONEP and Tech Specs have been addressed and/or at the direction of the lead evaluator, **trigger Event 4** to simulate an inadvertent initiation of RCIC.

B. The crew will take actions per the Loss of Feedwater Heating ONEP by reducing Core flow to 70Mlbm/hr.

C. The CRS will direct the RO to secure RCIC using Hard Card, 04-1-01-E51-1, Att. VI.

D. If sent to investigate inadvertent initiation, as I&C Maintenance wait 3 minutes then report problem is failed trip unit and a work order is required.

E. The CRS will determine that TS 3.5.3 Condition A applies.

Feedwater Master Level Controller failure:

A. When the CRS has addressed TS and at the direction of the lead evaluator, trigger **Event 5** to cause the feedwater master level controller to fail upscale.

B. Crew will take immediate actions in accordance with 05-1-02-V-7, Feedwater System Malfunctions ONEP by placing controller in Manual and controlling reactor water level.

C. If sent to investigate, as I&C Maintenance wait 3 minutes then report problem is unknown and a work order is required.

Spurious Main Turbine trip:

A. When the CRS has addressed subsequent actions of the ONEP and/or at the direction of the lead evaluator, trigger **Event 6** to cause Spurious Main Turbine Trip.

B. The crew will recognize a Main Turbine trip and Reactor Scram signal.

C. The crew will take actions of the Main Turbine Trip and Reactor Scram ONEPs.

D. The CRS will enter EP-2

ATWS

A. The crew will recognize control rods will fail to fully insert due to hydraulic block **(Event 7)**.

B. The CRS will enter EP-2A.

Standby Liquid Control Failure:

A. When the crew initiates Standby Liquid Control System the system will fail to inject. **(Event 8)**

Feedwater piping 'B' will rupture inside the drywell with inability to isolate.:

A. Five minutes after the scram signal, Feedwater Line B will rupture in the Drywell **(Event 9)**. ATC secures all condensate and feedwater. B21-F065B will fail to close to isolate the leak, this will prevent using Feedwater and Condensate to feed the reactor.

Termination:

A. Once rod movement has occurred and reactor water level is being controlled in band or as directed by Lead Evaluator:
   - Take the simulator to Freeze and turn horns off.
   - Stop and save the SBT report and any other recording devices.
   - Instruct the crew to not erase any markings or talk about the scenario until after follow-up questions are asked.

Critical Task		
Number	Description	Basis
1	* **When EP-2A requires Emergency Depressurization, the crew terminates and prevents all injection except boron, CRD, and RCIC prior to Emergency Depressurization.**	When emergency depressurization is required under ATWS conditions, control of RPV water level and RPV pressure must be carefully coordinated to avoid potentially damaging reactivity transients. Injection into the RPV is temporarily terminated to prevent uncontrolled injection of large amounts of cold water as RPV pressure decreases. After the SRVs are opened, control of RPV water level returns to the Level branch at #10, where injection is restored. (per 02-S-01-40, EP Technical Bases)
2	* **When it is determined that reactor water level cannot be restored and maintained above -191", the crew opens at least seven SRVs prior to exiting EP-2A.**	If RPV water level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level, emergency RPV depressurization is performed to maximize injection flow and minimize break flow. The Minimum Steam Cooling RPV Water Level is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Maintaining RPV water level above the Minimum Steam Cooling RPV Water Level thus ensures that the core remains adequately cooled. (per 02-S-01-40, EP Technical Bases)
3	* **After reactor pressure decreases to MSCP, the Crew commences and slowly raises injection to restore and maintain RPV level to greater than -191" prior to exiting EP-2A.**	"Minimum Steam Cooling Pressures," defined to be the lowest RPV pressures at which steam flow through open SRVs is sufficient to preclude the clad temperature of the hottest fuel rod from exceeding 1500°F even if the reactor core is not completely covered. When RPV pressure drops below the Minimum Steam Cooling Pressure, steam flow may no longer be sufficient to cool uncovered nodes and injection into the RPV must be reestablished. (per 02-S-01-40, EP Technical Bases)

4	* **Start SLC pumps prior to exceeding 110°F Suppression Pool temperature.**	The specified temperature of 110°F is the most limiting value of the Boron Injection Initiation Temperature, defined to be the greater of: • The highest suppression pool temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight of boron before suppression pool temperature exceeds the HCTL. • The suppression pool temperature at which a reactor scram is required by plant Technical Specifications. (per 02-S-01-40, EP Technical Bases)
5	* **Following an ATWS, insert control rods by manual scram and/or normal rod insertion prior to exiting EP-2A.**	Positive confirmation that the reactor will remain shutdown under all conditions is best obtained by verifying that all control rods are inserted to or beyond position 02. Position 02 is the “Maximum Subcritical Banked Withdrawal Position,” defined to be the greatest banked rod position at which the reactor will remain shutdown under all conditions. . (per 02-S-01-40, EP Technical Bases)
*	**Critical Task** (As defined in NUREG 1021 Appendix D)	

Facility: Grand Gulf Nuclear Station Scenario No.: 2 Op-Test No.: NRC LOT 2015

Examiners: ______________________ Operators: ______________________

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

1. Lower reactor power using Recirc Flow Control.
2. Start RHR A in Suppression Pool Cooling
3. Respond to a trip of CRD pump A.
4. Respond to CRD Accumulator Faults.
5. Respond to a Suppression pool leak in RHR pump A suction piping while operating in Suppression Pool Cooling.
6. Respond to both Heater Drain Pump Trip.
7. Respond to Recirc Line break
8. Respond to a Hotwell low level signal.
9. Respond to a RHR ‘B’ and ‘C’ system initiation failure.

Initial Conditions: Operating at 100% power.

Inoperable Equipment: None

Turnover:

The plant is at 100%. Planned activities for this shift are:

- Lower power to 90% (1300 Mwe) per IOI.
- Start RHR ‘A’ in Suppression Pool Cooling mode.

HPCS is out of service and EOOS is GREEN. It is a division 3 work week.

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 60 minutes

Event No.	Malf. No.	Event Type †	Event Description
1		N (BOP) R (ATC)	Lower Reactor power to 90% using FCV's (IOI 03-1-01-2 Attachment VIII)
2		N (BOP) TS (CRS)	Start RHR 'A' in Suppression Pool Cooling (04-1-01-E12-1, Tech Spec 3.5.1)
3	c11028a	C (BOP) A (CREW)	Respond to a Trip of CRD pump 'A' (CRD Malfunction ONEP, 05-1-02-IV-4)
4	z024_024_32_17 z024_024_36_21	C (ATC) TS (CRS) A (CREW)	Respond to two HCU Accumulator low pressure fault (SOI 04-1-01-C11-1; ARI 04-1-03-P680-4A2-D4, Tech Specs 3.1.5 Cond. A)
5	ct218a	C (BOP) TS (CRS)	Respond to Suppression pool leak from RHR 'A' suction line. (EP-3 & 4, EN-OP-115, Conduct of Operations, Tech Specs 3.6.2.2 Low suppression pool level Cond A and 3.5.1 Cond. C)
6	fw231a & b	C (ATC) A (CREW) R (ATC)	Respond to both Heater Drain Pumps trip, (Loss of Feedwater Heating 05-1-02-V-5)
7	rr063a	M (CREW)	Reactor Recirc Line 'A' break (EP-2)
8	rr040b	I (CREW)	Respond to a RHR 'B' and 'C' system initiation failure
9	p680_2a_e_9 fw115a, b & c	I (CREW)	Respond to a false Hotwell low level signal
† (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec			

Quantitative Attributes Table			
Normal Events	2	Abnormal Events	3
Reactivity Manipulations	2	Total Malfunctions	6
Instrument/Component Failures	6	EP Entries (Requiring substantive action)	1
Major Transients	1	EP Contingencies	3
Tech Spec Calls	3	Critical Tasks	3

SCENARIO ACTIVITIES:

Lower Reactor power to 90% using FCV's

A. If necessary, call the control room as the Shift Manager and direct the CRS to lower reactor power to 90% using IOI. All notifications have been made.

1. The crew will lower power using 03-1-01-2, Power Operations IOI, Attachment III, Power Reduction from Full Reactor Power to 60%. **(Event 1)**

Start RHR 'A' in Suppression Pool Cooling

A. After Reactor Power has been reduced to 90% contact the CRS to place RHR 'A' in Suppression Pool Cooling beginning with step 5.2.2 a (3). **(Event 2)**

B. The CRS will determine that Tech Spec 3.5.1 Condition A applies.

Trip of CRD pump 'A'

A. After Suppression pool cooling is in service and/or at the direction of the lead evaluator **trigger Event 3** to cause CRD pump 'A' to trip.

B. Crew will enter CRD Malfunction ONEP and perform immediate actions to start standby CRD Pump.

C. If asked to investigate pump trip:

1. Wait 3 minutes and respond as a Plant Operator and report "no apparent reason for pump trip and motor is hot to the touch

2. Wait 3 minutes and respond as Electrical Maintenance and report breaker 152-1505 186 device is tripped with Instantaneously flags on all three phases.

Two HCU Accumulator low pressure faults.

A. During the CRS pump trip of event 3 two HCU accumulators will indicate low pressure **(Event 4)**

1. One accumulator will clear when the standby pump is started.

2. If asked to provide local accumulator pressure, wait 5 minutes and report as plant operator that HCU 32-17 is 1620 psig and HCU 36-21 is 1510 psig.

B. The CRS will determine that Tech Spec 3.1.5 Condition A applies.

Suppression Pool Leak from RHR 'A' suction line.

A. After Immediate and Subsequent actions of CRD Malfunction ONEP and Tech Specs have been addressed and/or at the direction of the lead evaluator **trigger Event 5** to cause a leak on the suction line of RHR 'A' system.

B. A leak will develop on the down stream side of the RHR MOV suction valve E12-F004A.

C. Crew should trip RHR 'A' pump and close suction valve E12-F004A to isolate the leak.

   1. After E12-F004A is closed the leak will stop.

D. The CRS should enter EP-3 due to suppression pool water level below 18.34 FT.

E. The CRS should enter EP-4 due to Hi-Hi RHR 'A' room sump and RHR 'A' room flooded.

F. The CRS will determine that Tech Spec 3.6.2.2 Condition A applies due to low suppression pool level and 3.5.1 Condition C now applies due to loss of HPCS and RHR 'A'.

Both Heater Drain Pumps Trip

A. After EPs and Tech Specs have been addressed and/or at the direction of the lead evaluator **trigger Event 6** to cause both Heater Drain Pumps to trip.

B. Crew will enter Loss of Feedwater Heating and Feedwater System Malfunctions ONEPs.

C. Crew will reduce core flow to 70 Mlbm/hr per immediate operator actions.

Reactor Recirculation system 'A' line break

A. After all immediate and subsequent actions of ONEPS have been addressed and/or at the direction of the lead evaluator **trigger Event 7** to cause Recirc system 'A' line break.

B. Crew will enter EP-2, EP-3, Turbine Trip and Reactor Scram ONEPs.

RHR ‘B’ and ‘C’ System Initiation Failure

A. The crew will recognize and respond to a fail to initiate on RHR ‘B’ and ‘C’ **(Event 8)**

B. The crew will manually initiate RHR ‘B/RHR ‘C’ systems.

False Hotwell Level Low

A. Two minutes after Mode Switch is taken to SHUTDOWN, a false Hotwell level low will occur **(Event 9)** causing a trip of all Condensate pump which will cause a trip of all Condensate Booster pumps and Feedwater pumps.

B. This failure will cause a loss of all high pressure feed systems. (Recirc leak will be greater than RCIC system flow).

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

D. Crew will restore reactor water level with LPCS, RHR ‘B’ and ‘C’ (if manually initiated)

Critical Task		
Number	Description	Basis
1	* **Open at least 7 SRVs when RPV water level is between -160" and -190".**	If an injection source is available but the decreasing RPV water level trend cannot be reversed before RPV water level drops to the Minimum Steam Cooling RPV Water Level (-191 in.), emergency RPV depressurization is required to permit injection from low head systems, maximize flow from available injection sources, and minimize the flow through any primary system break. (per 02-S-01-40, EP Technical Bases)
2	* **After Emergency Depressurization, restore and maintain RPV level above -191" using available injection systems prior to exiting EP-2.**	The Minimum Steam Cooling RPV Water Level is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. (per 02-S-01-40, EP Technical Bases)
3	* **When ECCS fails to initiate, the crew manually initiates ECCS prior to reactor pressure going below 300 psig.**	Take manual actions (in accordance with procedure direction, if available) when automatic actions do not occur. (per EN-OP-120, Operator Fundamentals Program). 300 psig is bases on maximum discharge pressure for RHR LPCI systems B and C injection is 285 psig.
* **Critical Task** (As defined in NUREG 1021 Appendix D)		

Facility: Grand Gulf Nuclear Station Scenario No.: 3 Op-Test No.: NRC LOT 2015

Examiners: ______________ Operators: ______________

______________ ______________

______________ ______________

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

1. Start a Condensate Pump.
2. Withdraw Control Rods to Establish Main Turbine BCV's 10% open.
3. RCIC Steam line break with failure to auto isolate.
4. One IRM channel fails upscale.
5. Loss of ESF Transformer 21.
6. Startup Level Control Valve Fails Open.
7. Steam Line Break in Drywell/Failure of automatic scram
8. Loss of all level indication.

Initial Conditions:

- Reactor startup in progress.
- Reactor pressure is 400 psig
- Reactor power is 5%

Inoperable Equipment: None

Turnover:

- A reactor startup is in progress.
  - Step 84 of Control Rod Movement Sequence is complete
  - SJAE 'B' is in warm up 04-01-N62-1 step 4.2.2r
  - Step 30 of Attachment XV in 03-1-01-1
- The Condensate system is lined up as follows:
  - CFFF is in service
  - Precoat Filters are not in service
  - 4 Deepbed demins are in service
- Start a Condensate pump after turnover.
- Withdraw Control Rods to obtain 10% Bypass valve position on all valves

Scenario Notes:

This scenario is a modified version of the 2014 NRC Exam Scenario 5 (spare scenario, not used).

Validation Time (60-90 min): 70 min

Event No.	Malf. No.	Event Type †	Event Description
1		N (BOP/CRS)	Start a Condensate Pump (04-1-01-N19-1 Condensate System)
2		R (ATC/CRS)	Withdraw Control Rods to Establish Main Turbine BCV's 10% open (04-1-01-C11-2 Rod Control and Information System)
3	e51050 Att. 3	C (BOP) A (CREW) TS (CRS)	RCIC Steam line break with failure to auto isolate. (EP-4, TS 3.5.1)
4	c51004b	I (ATC)	IRM channel 'B' fails upscale
5	r21180 r21218 (rm)	C (BOP/CRS) A (CREW) TS (CRS)	Loss of ESF Transformer 21 (05-1-02-I-4, Loss of AC Power) Division 2 LSS Failure (TS 3.8.1)
6	fw274	C (ATC/CRS) A (CREW)	Startup Level Control Valve Fails Open (05-1-02-V-7, Feedwater System Malfunctions)
7	ms065a c71076	M (CREW)	Steam Line Break in Drywell with failure of automatic scram (EP-2, 3).
8	rr188a rr188b rr188c rr188d rr188e rr188f rr188g rr188h rr188i ltb21n044d_a ltb21n044d_a ao_c34r614_g	I (CREW)	Loss of all Reactor water level indication (EP-5)
†	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec		

Quantitative Attributes Table

Normal Events	1	Abnormal Events	3
Reactivity Manipulations	1	Total Malfunctions	6
Instrument/Component Failures	5	EP Entries (Requiring substantive action)	2
Major Transients	1	EP Contingencies	2
Tech Spec Calls	2	Critical Tasks	3

SCENARIO ACTIVITIES:

Start a Condensate Pump (Normal):

A. After turnover the crew will start a second Condensate Pump per 04-1-01-N19-1 and 03-1-01-1, IOI.
   1. The crew (BOP) will start a second Condensate pump. **(Event 1)**

Withdraw Control Rods to Establish Main Turbine BCV's 10% open

A. After the second Condensate pump is started, the crew (ATC) will withdraw control rods per movement sheet and 04-1-01-C11-2, Rod Control and Information System, to obtain at least 10% bypass valve position on all valves. **(Event 2)**

RCIC Steam line break with failure to auto isolate

A. After Bypass control valves indicate 10% and/or at the direction of the lead evaluator, **trigger Event 3** to cause a RCIC steam line break.
B. RCIC will fail to isolate, E51-F063 and F064 will not close automatically.
C. The crew (BOP) will take the actions per EP-4 and EN-OP-120.
D. The crew will manually isolate RCIC by closing E51-F063 and F064.
E. The CRS will determine that TS 3.5.3 Condition A applies.

IRM channel 'B' fails upscale

A. When the CRS has addressed TS and at the direction of the lead evaluator, **trigger Event 4** to cause IRM channel 'B' to fail upscale.
B. The crew (ATC) will take action per ARI 04-1-02-1H13-P680-7A-A8, to bypass the IRM and reset the ½ scram.

Loss of ESF Transformer 21

A. After the IRM is bypassed and the ½ scram is reset and at the direction of the lead evaluator, **trigger Event 5** to cause ESF transformer to trip.
B. The crew (BOP) will recognize a loss of power to buses 16AB and !&AC. The 17AC bus will be re-energized by its emergency diesel generator. The 16AB bus will experience a LSS failure.

C. The crew (BOP) will respond by taking immediate action per the Loss of AC Power ONEP, 05-1-01-I-4, by re-energizing 16AB from an alternate feeder.

D. The crew will re-open the Containment isolation valves for instrument air and enter the Automatic Isolations ONEP.

E. The CRS will determine that TS 3.8.1 Condition F applies.

<u>Startup Level Control Valve Fails Open</u>

A. After immediate, subsequent actions of all associated ONEPs and Tech Specs have been addressed and at the direction of the lead evaluator, **trigger Event 6** to cause the feedwater Startup Level Control Valve to fail open.

B. The crew will recognize a rising level and power.

C. The crew (ATC) will place the Startup Level Control valve in Manual and control level per Feedwater Malfunctions ONEP, 05-1-01-V-7, immediate actions.

<u>Steam Line Break in Drywell with failure of automatic scram</u>

A. After immediate, subsequent actions of all associated ONEPs have been addressed and at the direction of the lead evaluator, **trigger Event 7** to cause a main steam line to break in the Drywell.

B. The crew will recognize rising drywell pressure, an automatic scram will not work and the crew will be required to insert a manual scram or manually place the mode switch to shutdown.

C. The CRS will enter EP-2

<u>Loss of all Reactor water level indication</u>

A. Approximately 5 minutes after scram all level indication will fail upscale due to elevated drywell temperatures. **(Event 8)**

B. The CRS will enter EP-5

Termination:

A. Once the RPV is verified to be flooded or as directed by Lead Evaluator:

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

Critical Task		
Number	Description	Basis
1	* **When Drywell pressure exceeds scram setpoint, place the Mode Switch to shutdown prior to exceeding 2.5 psig Drywell pressure.**	**Licensed operators SHALL immediately insert a manual scram whenever any of the following conditions occurs:** • **Operating parameters exceed any of the reactor protection set points and an automatic shutdown does not occur. (EN-OP-115, 5.2)**
2	* **When reactor water level cannot be determined open at least seven SRVs prior to flooding the RPV to the Main Steam Lines.**	**The EP-5 flooding strategy depressurizes the RPV, then controls injection to fill the RPV to the elevation of the main steam lines. This provides the only positive method of assuring adequate core cooling when RPV water level cannot be determined. (02-S-01-40 Att VIII, 9.1)**
3	* **When reactor water level cannot be determined, inject with one or more available systems and observe RPV flooded to the MSLs prior to securing injection systems.**	**Flooding the RPV to the elevation of the main steam lines provides the only positive method of assuring adequate core cooling when RPV water level cannot be determined. (02-S-01-40 Att VIII, 9.1)**
* **Critical Task** (As defined in NUREG 1021 Appendix D)		