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

Tier	Group	RO K/A Category Points K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	SRO-Only Points A2	G*	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	RO 1	RO 2	RO 3	RO 4	RO 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	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				**Knowledge of the reasons for the following responses as they apply to a partial or complete loss of A.C. power:** AK3.05: Reactor Scram CFR: 41.5	3.7	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 operate 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 Form ES-401-1 Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / SRO)									
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3									
295007 High Reactor Pressure / 3									
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2	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 Form ES-401-1
Plant Systems - Tier 2/Group 1 (RO / 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					**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:** K1.06: Reactor vessel 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:** G2.4.11 Knowledge of abnormal condition procedures. 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	295003 AK3.05	De-selected due to similarity to question #46.
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	215005 K1.06	De-selected, N/A for GGNS
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.
2/1	400000 G2.4.11	De-selected due to Low Operational value for discriminatory RO level question
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 Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (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	#
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:** G2.2.38: Knowledge of conditions and limitations in the facility license. CFR: 43.5	4.5	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									
295011 High Containment Temp / 5					X		**Ability to determine and/or interpret the following as they apply to high Containment Temperature:** AA2.02: Containment Pressure CFR: 43.5	4.0	83
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								X				**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:** A2.03: Stuck Open SRV	3.1	88
259002 Reactor Water Level Control														

261000 SGTS								X				261000 Standby Gas Treatment System **Ability to (a) predict the impacts of the following on the Standby Gas Treatment System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:** A2.05: Fan Trips	3.1	90
262001 AC Electrical Distribution														
262002 UPS (AC/DC)											X	**For the UPS:** G2.4.32: Knowledge of operator response to loss of all annunciators. CFR: 43.5	4.0	89
263000 DC Electrical Distribution														
264000 EDGs														
300000 Instrument Air														
400000 Component Cooling Water														
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	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														
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														
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								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:** A2.01: Interlock Failure CFR: 41.5	3.6	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	**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	295021 2.2.38	De-selected due to low sampling of CFR 55.43
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.
2/1	239002 A2.04	De-selected due to inability to write discriminatory SRO level question for this K/A.
2/1	261000 A2.05	De-selected due to overlap with operating examination.
2/2	202002 A 2.04	De-selected, N/A for GGNS.
2/2	234000 A2.01	De-selected due to low sampling of CFR 55.43
1/2	295010 AA2.06	De-selected due to inability to write discriminatory SRO level question for this K/A.
2/1	262002 2.4.11	Had duplicate concept with question 78 due to low number of AOPs at GG so NRC selected 2.4.32 and wrote new question on loss of annunciators.

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 changes to procedures. 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		
	2.4.44	2.4.44 Knowledge of emergency plan protective action recommendations 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** Examination Level: RO ☒ SRO ☐		Date of Examination: 12/7/2015 Operating Test Number: LOT-2015
Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations AR1	R - M	Loss of Shutdown Cooling, Time to 200F Determination GJPM-OPS-2015-AR1
Conduct of Operations		
Equipment Control AR3	R - N	Electrical Print Reading (Determine effect of removing fuses in RPS system) GJPM-OPS-2015-AR3
Radiation Control AR2	R-N	Exposure Limits GJPM-OPS-2015AR2
Emergency Plan AR4	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 AS1	R - D	Determine Firewatch Requirements GJPM-OPS-2015-AS1
Conduct of Operations AS2	R - N	Manual On-Line Risk Assessment GJPM-OPS-2015-AS2
Equipment Control AS3	R – D	Tagout Removal Approval GJPM-OPS-2015-AS3
Radiation Control AS4	R - D	Review Liquid Radwaste Discharge Permit GJPM-OPS-2015-AS4
Emergency Plan AS5	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-2015S1 **(S1)**	A - D - S	1
b.	259001 A3.10 (3.4/3.4), Defeat Feed Pump Level 9 Trips GJPM-OPS-2015CR2 **(CR2)**	D - C - L	2
c.	239001 A2.11 (4.1/4.3), Slow Closing MSIV GJPM-OPS-2015S3 **(S3)**	A - N - S	3
d.	209002 A1.01 (3.6/3.7), Performing HPCS Quarterly Functional Test, GJPM-OPS-S4 **(S4)**	A - D – EN - S	4
e.	223001 A4.06 (4.0/4.0), EP-1 Attachment 14, Containment Venting GJPM-OPS-2015S5 **(S5)**	C - D - E - L	5
f.	212000 A4.02 (3.6/3.7), Reactor Manual Scram Switch Test, GJPM-OPS-2015S6 **(S6)**	A - D - S	7
g.	400000 A4.01 (3.1/3.0), Manual Start of SSW ‘A’, GJPM-OPS-2015S7 **(S7)**	D - S	8
h.	261000: A4.03 – 3.0, Secure SGTS A Train, GJPM-OPS-2015S8 **(S8)**	N - S	9
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)			
i.	262001 A2.11 (3.2/3.6), Reset Undervoltage Lockouts on BOP Buses, GJPM-OPS-2015PS-6 **(P1)**	D - R - L	6
j.	295016 A1.07 (4.2/4.3), Perform Attachment III of Shutdown From Remote Shutdown panel ONEP, GJPM-OPS-2015PS-7 **(P2)**	D - E – EN - L	7
k.	2.1.30: (4.4/4.0), Manually Initiate Fire Protection to the ‘B’ RPS Motor Generator Room, GJPM-OPS-2015PS8 **(P3)**	A – E - N	8
*	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-2015S1 **(S1)**	A - D - S	1
b.	259001 A3.10 (3.4/3.4), Defeat Feed Pump Level 9 Trips GJPM-OPS-2015CR2 **(CR2)**	D - C - L	2
c.	239001 A2.11 (4.1/4.3), Slow Closing MSIV GJPM-OPS-2015S3 **(S3)**	A - N - S	3
d.	209002 A1.01 (3.6/3.7), Performing HPCS Quarterly Functional Test, GJPM-OPS-S4 **(S4)**	A - D – EN - S	4
e.	223001 A4.06 (4.0/4.0), EP-1 Attachment 14, Containment Venting GJPM-OPS-2015S5 **(S5)**	C - D - E - L	5
f.	212000 A4.02 (3.6/3.7), Reactor Manual Scram Switch Test, GJPM-OPS-2015S6 **(S6)**	A - D - S	7
g.	400000 A4.01 (3.1/3.0), Manual Start of SSW ‘A’, GJPM-OPS-2015S7 **(S7)**	D - S	8
h.	N/A		
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)			
i.	262001 A2.11 (3.2/3.6), Reset Undervoltage Lockouts on BOP Buses, GJPM-OPS-2015PS-6 **(P1)**	D - R - L	6
j.	295016 A1.07 (4.2/4.3), Perform Attachment III of Shutdown From Remote Shutdown panel ONEP, GJPM-OPS-2015PS-7 **(P2)**	D - E – EN - L	7
k.	2.1.30: (4.4/4.0), Manually Initiate Fire Protection to the ‘B’ RPS Motor Generator Room, GJPM-OPS-2015PS8 **(P3)**	A – E - N	8

* 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-2015S1 **(S1)**	A - D - S	1
b. N/A		
c. 239001 A2.11 (4.1/4.3), Slow Closing MSIV GJPM-OPS-2015S3 **(S3)**	A - N - S	3
d. 209002 A1.01 (3.6/3.7), Performing HPCS Quarterly Functional Test, GJPM-OPS-S4 **(S4)**	A - D – EN - S	4
e. N/A		
f. N/A		
g. N/A		
h. N/A		
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
i. 262001 A2.11 (3.2/3.6), Reset Undervoltage Lockouts on BOP Buses, GJPM-OPS-2015PS1 **(P1)**	D - R - L	6
j. N/A		
k. 2.1.30: (4.4/4.0), Manually Initiate Fire Protection to the 'B' RPS Motor Generator Room, GJPM-OPS-2015PS8 **(P3)**	A – E - N	8

* 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 Scenario No.: 1 (Spare) 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. Rotate Power to bus 17AC from ESF 12 to ESF 21.
2. Respond to a loss 17AC with a Diesel Generator failure.
3. Loss of EPA breaker on 'B' RPS.
4. Trip of the 'B' RPS Alternate Power Supply with Inadvertent SCRAM of 4 Control Rods
5. Respond to low Main Turbine Lube Oil Pressure and Main Turbine Trip and Reactor Scram
6. Take actions for an ATWS.
7. RCIC Steam line break with failure to auto isolate with manual isolation available.
8. Respond to a Feedwater pump trip.

Initial Conditions: Operating at 95% power.

Inoperable Equipment: None

Turnover:

The plant is at 95% following a sequence exchange. SSW A is operating in preparation for weekly chemical addition. Planned activities for this shift are:

- 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 1 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)	Rotate Power to 17AC from ESF 12 to ESF 21.(SOI 04-1-01-R21-17)
2	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 condition B)
3	c71077b	C (ATC) C (BOP) A (CREW) TS (CRS)	Loss of EPA breaker on RPS 'B' (Loss of One or Both RPS Buses, 05-1-02-III-2) TR 3.1.5 Condition A (until half scram is reset)
4	LO_1C71M607B DI_1C71M607B ZO25025_40_33 ZO25025_40-25 ZO25025_32-37 ZO25025_32-21	C (ATC) R (ATC) A (CREW) TS (CRS)	Trip of the 'B' RPS Alternate Power Supply with Inadvertent SCRAM of 4 Control Rods (Control Rod/Drive Malfunctions ONEP, 05-1-02-IV-1; Reduction in Recirculation Flow Rate ONEP, 05-1-02-III-3, Tech Spec TR 3.1.5 Condition A)
5	tc093 AO_1N34R600 DI_1N34M661 DI_1N34M662 p680_10a_c_1	M/A (CREW)	Respond to Low Main Turbine Lube Oil Pressure (ARI 1H13-P680-10A-C2) Respond to Reactor Scram and/or Main Turbine Trip (Reactor Scram and Turbine Generator Trip ONEPS, EP-2)
6	c11164	M (CREW)	ATWS (EP-2A)
7	e51050 Att. 3	C (BOP)	RCIC steam line break with failure to auto isolate, manual isolation available. (EN-OP-120, 02-S-01-43, EP-4)
8	fw123a (b)	C (ATC)	Feedwater pump A (B) trip
†	(N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec		

* Repeated task from last 2 NRC exams.

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

SCENARIO ACTIVITIES:

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

A. After turnover contact CRS to perform power swap for Bus 17AC. **(Event 1)**

B. Using 04-1-01-R21-17, SOI for 17AC bus, the BOP will rotate power from ESF Transformer 12 to normal source of ESF 21.

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 2)**

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 B4 applies and require surveillance 06-OP-1R21-W-0001 Attachment II.

Loss of EPA Breaker on 'B' RPS

NRC Q

A. When the crew has addressed all required Tech Specs and/or at the direction of the lead evaluator, **trigger Event 3** to cause EPA breaker C71-S003D to trip. *(This is different from previous exams due to the failure of Electrical Protection Assemblies the RPS M/G will continue to run.)*

B. The ATC will recognize and report a Division 2 half scram and determine a loss of 'B' RPS.

C. The BOP will respond to the back panel area and determine a loss of power to the 'B' RPS.

D. If sent to investigate loss of RPS 'B', wait 3 minutes and report C71-S003D has tripped on undervoltage.

E. CRS will enter the Loss of One or Both RPS Buses ONEP to transfer power to the alternate source.

F. The BOP will transfer 'B' RPS power to alternate source

G. The ATC will reset half scram

H. The CRS will determine that TR 3.1.5 Condition A applies as long as the half scram is present.

Trip of the 'B' RPS Alternate Power Supply with Inadvertent SCRAM of 4 Control Rods

A. When the crew has addressed all required Tech Specs and the subsequent actions of the Loss of One or Both RPS Buses ONEP and/or at the direction of the lead evaluator, **trigger Event 4** to cause a trip 'B' RPS Alternate Power Supply.

B. The ATC will recognize and report a Division 2 half scram and determine a loss of 'B' RPS, also recognize and report 4 Control Rods have SCRAMMED.

C. The ATC will reduce core flow to 70 Mlbm/hr per 05-1-02-IV-1, Control Rod/Drive Malfunctions section 2.4.

D. The BOP will respond to the back panel area and determine a loss of power to the 'B' RPS.

E. The CRS will enter 05-1-02-IV-1, Control Rod/Drive Malfunctions ONEP and 05-1-02-III-3, Reduction in Recirc Flowrate ONEP and **Re-enter Tech Specs TR 3.1.5 Condition A.**

F. The ATC will plot current position on the Power to Flow map. The BOP will verify the plot.

G. If sent to investigate loss of RPS 'B', wait 3 minutes and report both EPA breakers for Alternate source, C71-S003H and S003F are tripped on undervoltage.

H. If sent to investigate alternate supply for RPS 'B' at 52-164227, report breaker is tripped free.

Low Main Turbine Lube Oil Pressure

A. When Reactor parameters have stabilized and/or at the direction of the lead evaluator, **trigger Event 5** to cause a reduction in Main Turbine Lube Oil Pressure.

B. The crew will recognize a reduction in Main Turbine Lube Oil pressure by alarms and indication.

Main Turbine Trip / Reactor Scram

A. Approximately 1 minute after Main Turbine Low Lube Oil pressure alarm the Main Turbine will trip.

B. The crew will recognize a Main Turbine Trip

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

D. The CRS will enter EP-2

ATWS, 15% (Will maintain >5% power)

A. The crew will recognize control rods will fail to fully insert due to hydraulic block **(Event 6)**. *(This is different than other previous used ATWS malfunctions due to severity is at 15%)*

B. The CRS will enter EP-2A.

RCIC steam line break with failure to auto isolate

A. At approximately Five minutes after the scram signal and/or at the direction of the lead evaluator, **Trigger Event 7,** RCIC Steam Line will break. *(This is different than other previous used malfunctions due to the leak can be manually isolated and steam leak stopped.)*

B. The BOP will recognize and report indications of a RCIC steam line break and entry into EP-4.

C. The BOP will recognize failure to auto isolate

D. The BOP will manually isolate the RCIC system by closing E51-F063 and F064.

E. The CRS will enter EP-4.

Feedwater Pump trip (which ever feedwater pump is operating):

A. At approximately Five minutes after the scram signal and/or at the direction of the lead evaluator, **Trigger Event 8 RFPT 'A' /** Event 9 RFPT 'B', Feedwater pump will trip.

B. ATC recognizes and restarts the standby feedwater pump and maintains within level band.

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	∗ **Per step L-7 terminate and prevent all RPV injection except for Boron injection, CRD and RCIC until level reaches -70" wide range prior to THI. (i.e. >10% power swings peak to peak on APRM indication)**	Water level must be lowered to reduce subcooling and prevent instabilities.
2	∗ **Start standby Reactor Feed Pump prior to Emergency Depressurization.**	If not performed properly would result is direct adverse consequences or significant degradation in the mitigative capability of the plant. Correct performance prevents degradation of any barrier to fission product release (i.e. an emergency depressurization will be required)
3	∗ **Isolate all systems (RCIC) discharging outside the primary CTMT through a break, except systems needed for fire suppression or EP actions prior to the Main Steam Tunnel exceeding 250°F, its Max safe value in table 10 of EP-4.**	If a RCIC steam leak with failure to isolate occurs, then RCIC should be shutdown and isolated unless required to assure adequate core cooling. This is to prevent an unmonitored release and loss of fission product barrier.
4	∗ **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. Start RHR A in Suppression Pool Cooling
2. Respond to a trip of RHR 'C' Jockey pump
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 Reactor Instrument Line Failure
8. Respond to Logic Power Failure to LPCS/RHR A
9. Division 2 ECCS High Drywell Pressure fail to Initiate
10. Respond to a Hotwell low level signal.

Initial Conditions: Operating at 100% power.

Inoperable Equipment: None

Turnover:

The plant is at 100%.

- RCIC is tagged for maintenance. LCO 3.5.3 Condition A has been entered, HPCS is verified OPERABLE.

Planned activities for this shift are:

- Start RHR 'A' in Suppression Pool Cooling mode
- SSW A is operating in preparation for weekly chemical addition.

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 60 minutes

Event No.	Malf. No.	Event Type †	Event Description
1*		N (BOP) TS (CRS)	Start RHR 'A' in Suppression Pool Cooling (04-1-01-E12-1, Tech Spec 3.5.1 Condition A, TR 6.8.2 Condition A)
2	DI_1E12M601C	TS (CRS)	RHR 'C' Jockey Pump Trip (Tech Spec 3.5.1 Condition C, 3.3.6.4 Condition C)
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) A (CREW) TS (CRS)	Respond to two HCU Accumulator low pressure fault (SOI 04-1-01-C11-1; ARI 04-1-03-P680-4A2-D4) Tech Spec TR 3.1.5 Condition A (for the rod that is in alarm)
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 Spec 3.6.2.2 Low suppression pool level Cond A, 3.6.1.7 Condition A, 3.6.2.3 Condition A)
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; Reduction in Recirculation System Flow Rate ONEP, 05-1-02-III-3)
7	rr062	M (CREW)	Respond to Reactor Instrument Line Failure (EP-2)
8	r21219 r21220	I (CREW)	Respond to Logic Power Failure to LPCS/RHR 'A' (Alarm Response Instruction, 04-1-03-P601-21A-H8, LPCS OOSVC)
9	rr040f rr041f	I (CREW)	Division 2 ECCS High Drywell Pressure fails to initiate, (EN-OP-120).
10	p680_2a_e_9 fw115a, b & c	I (CREW)	Respond to a false Hotwell low level signal (EP-2, Alarm Response Instruction 04-1-03-P680-2A-E9, CNDSR HTWL LVL LO)
† (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec			

* Repeated task from last 2 NRC exams.

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

SCENARIO ACTIVITIES:

Start RHR ‘A’ in Suppression Pool Cooling

A. After Crew has taken shift contact the CRS to place RHR ‘A’ in Suppression Pool Cooling beginning with step 5.2.2 a. **(Event 1)**

B. When Test Return to Supp Pool Valve E21-F024A is opened the CRS will determine that Tech Spec 3.5.1 Condition A applies and TR 6.8.2 condition A applies.

RHR ‘C’ Jockey Pump Trip

A. After Crew has placed suppression pool cooling in service, Tech Specs have been addressed and/or at the direction of the lead evaluator **trigger Event 2** to cause RHR ‘C’ Jockey Pump breaker (52-161135) to trip.

B. Crew will respond using Alarm Response Instructions.

C. If asked to investigate pump trip, wait 3 minutes and report breaker 52-161135 is in the trip free condition.

D. CRS will declare RHR ‘C’ INOP and determine Tech Spec 3.5.1 Condition C applies.

E. Crew will rackout the breaker for RHR ‘C’ pump, 152-1609, to protect the piping.

Trip of CRD pump ‘A’

A. After Tech Specs have been addressed and RHR ‘C’ pump breaker is racked out 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 has a 186 device tripped with Instantaneous flags on all three phases.

Two HCU Accumulator low pressure faults.

A. During the CRS pump trip of event 2 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 1580 psig and HCU 36-21 is 1610 psig.

B. The CRS will instruct to recharge HCU 32-17 per ARI and SOI, 04-1-01-C11-1.

C. No Tech Spec is required for accumulator due to >1520 psig, however, TR 3.1.5 condition A for the one rod that is in alarm.

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 downstream 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, 3.6.1.7 Condition A applies due Containment Spray ‘A’ INOP, 3.6.2.3 Condition A applies due to Suppression Pool Cooling ‘A’ INOP and 3.6.1.3 Condition A due to PCIV for ‘A’ RHR system being INOP. Tech Spec on 3.5.1 due to RHR ‘A’ was already INOP.

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, Feedwater System Malfunctions, and Reduction in Recirculation System Flow Rate ONEPs.

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

Reactor Instrument Line Failure

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 Reactor Below Core Plate pressure instrument line break. Drywell pressure will slowly rise causing the crew to take action to place the mode switch in SHUTDOWN.

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

C. The leak propagates into a Reactor Recirc line break that cause reactor level to reduce.

LPCS / RHR ‘A’ Logic Power Failure

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

B. The crew will manually start the LPCS pump (per ARI P601-21A-H8, LPCS SYS OOSVC, step 4.6) and attempt to manually (locally) open the injection valve.

C. Injection valve will not manually open.

Division 2 ECCS High Drywell Pressure fail to initiate

A. One of the two drywell pressure transmitters for Division 2 ECCS to auto initiate fails low **(Event 9)**, Div 2 ECCS will not auto initiate requiring the crew to manually initiate the Division 2 ECCS system. (RHR B only, RHR C is unavailable due to jockey pump loss)

False Hotwell Level Low

A. Two minutes after Mode Switch is taken to SHUTDOWN, a false Hotwell level low will occur **(Event 10)** 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.

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

D. Crew will restore reactor water level with RHR ‘B’

Termination:

A. After Reactor water level is restored by RHR ‘B’ and Condensate systems and 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	∗ **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 by Arming and Depressing Div 2 ECCS Manual Initiation Pushbutton 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. Component Cooling Water Pump 'A' trip with Failure of Standby pump to Auto Start.
2. MCC 17B01 Trip, HPCS System INOP
3. IRM Channel 'A' fails upscale
4. Loss of Power to ESF 15AA, with Diesel Generator restoring Bus.
5. IRM Channel 'E' fails downscale
6. Condensate Booster Pump 'A' Trip.
7. Component Cooling Water Pump 'B' and 'C' trip
8. Mode Switch Failure to Scram.
9. Loss of Offsite Power with 15AA lockout / Small LOCA
10. Division 2 Diesel Generator Fails to Auto Energize 16AB.

Initial Conditions:

- Reactor startup in progress.
- Reactor pressure is 400 psig
- Reactor power is 4%

Inoperable Equipment: None

Turnover:

- A reactor startup is in progress.
  - Step 83 of Control Rod Movement Sequence is complete
  - SJAE 'B' is in warm up 04-01-N62-1 step 4.2.2r
  - Step 32 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
- Preps are being made to start the first Reactor Feedwater Pump
- Severe Weather is in the area, Tornado Watch is in affect

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	p42151a	C (BOP) A (CREW)	Component Cooling Water Pump 'A' Trip with Failure of Standby Pump to Auto Start (05-1-02-V-1, Loss of Component Cooling Water ONEP, EN-OP-120, 02-S-01-43)
2	R21142DD	TS (CRS)	MCC 17B01 Trip, HPCS System INOP (Tech Specs 3.5.1 Condition B, 3.8.7 Condition D.1, Tech Specs 3.7.2 Condition A.1, and Loss of AC Power ONEP, 05-1-02-I-4)
3*	c51004a	C(ATC)	IRM Channel 'A' fails upscale (ARI P680-5A-A8)
4	DI_1R21M606A	C (BOP) A (CREW)	Loss of Power to ESF Bus 15AA with Diesel Restoring Bus (Loss of AC Power, 05-1-02-I-4, Loss of Instrument Air, 05-1-02-V-9 and Automatic Isolations, 05-1-02-III-5 ONEPs)
5	c51005e	TS (CRS)	IRM Channel 'E' fails downscale (Tech Specs 3.3.1.1 and TRM 3.3.2.1)
6	fw118a	C (ATC) A (CREW)	Condensate Booster Pump Trip (05-1-02-V-7, Feedwater System Malfunctions ONEP)
7	p42151b p42151c	M (CREW)	Component Cooling Water pumps 'B' and 'C' trip (05-1-02-V-1, Loss of Component Cooling Water, 05-1-02-I-1, Reactor Scram; 05-1-02-I-2, Turbine and Generator Trips ONEP, EP-2)
8	DI_1C71M602	C (ATC)	Mode Switch Failure with failure to AUTO Scram, Manual Scram Available (EN-OP-115)
9	r21135 r21139e	M (CREW)	Loss of Offsite Power with ESF bus 15AA lockout (Loss of AC Power ONEP, 05-1-02-I-4) and Small LOCA (EP 2)
10	n41142b	C (BOP)	Division 2 EDG fails to auto energize ESF Bus 16AB (Loss of AC Power ONEP, 05-1-02-I-4)
† (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor, (A)bnormal (TS) Tech Spec			

* Repeated task from last 2 NRC exams.

Quantitative Attributes Table

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

SCENARIO ACTIVITIES:

Component Cooling Water pump 'A' trip

A. After turnover and/or at the direction of the lead evaluator, **trigger Event 1** to cause CCW pump 'A' to Trip.

B. Standby CCW pump 'B' will fail to auto start.

C. The crew (BOP) will take the actions per EN-OP-120 and 05-1-02-V-1, Loss of Component Cooling Water ONEP and manually start the standby pump.

D. The CRS will enter 05-1-02-V-1, Loss of Component Cooling Water ONEP.

MCC 17B01 trip

A. After immediate, subsequent actions of all associated ONEPs have been addressed and at the direction of the lead evaluator, **trigger Event 2** to cause a MCC 17B01 trip.

B. The crew will recognize loss of 17B01 due to loss of power to HPCS MOVs and HPCS service water system.

C. The CRS will enter Loss of AC Power ONEP, and determine that TS 3.8.7Condition D1, TS 3.7.2 Condition B, and 3.5.1 Condition B applies.

IRM Channel 'A' fails upscale

A. After Tech Specs have been addressed and at the direction of the lead evaluator, **trigger Event 3** to cause IRM channel 'A' to fail full upscale.

B. The crew will recognize a division 1 half scram and rod block.

C. The crew will perform action per ARI by bypassing the IRM channel and resetting the half scram condition.

D. The CRS will direct the bypassing of the IRM channel and the resetting of the half scram.

E. The CRS will determine that NO Tech Specs are entered at this time but will refer to Tech Specs 3.3.1.1 and 3.3.2.1. Three channels are required. CRS may enter a potential LCO.

Loss of Power to ESF bus 15AA with EDG Restoring Power:

A. When Tech Specs have been addressed and any briefs are complete, at the Lead Evaluator's discretion, **trigger Event 4** to cause incoming feeder breaker 152-1514 to trip removing power to 15AA.

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

C. The BOP will verify EDG powers bus 15AA and restore Instrument Air to the Containment by opening P53-F001 on P870 using 05-1-02-V-9, Loss of Instrument Air ONEP. (P53-F001 fails closed on loss of power and must be manually re-opened)

D. The CRS should also enter 05-1-02-III-5, Automatic Isolations ONEP and 05-1-02-III-1, Inadequate Decay Heat Removal ONEP (due to the loss of Fuel Pool Cooling and Cleanup).

E. The CRS will determine that Tech Spec 6.4.1, Continuous conductivity monitoring applies.

IRM Channel 'E' fails downscale

A. After Tech Specs have been addressed, Immediate and subsequent actions of ONEPs and at the direction of the lead evaluator, **trigger Event 5** to cause IRM channel 'E' to fail downscale.

B. The crew will recognize 'E' IRM in the downscale indication with a control rod block.

C. The crew will perform action per ARI, but cannot bypass the IRM channel due to one is already bypassed in that division.

D. The CRS will determine that Tech Specs 3.3.1.1 and 3.3.2.1 apply.

Condensate Booster Pump 'A' trip

A. After all Tech Spec actions has been address and/or at the direction of the lead evaluator, **trigger Event 6** to cause Condensate Booster Pump 'A' to trip.

B. The crew (ATC) will take action per 05-1-02-V-7, Feedwater System Malfunctions ONEP and manually start the 'B' or 'C' Condensate Booster Pump

NOTE: If the standby condensate booster pump is not started in approximately 30 seconds, reactor water level will lower below the scram setpoint (+11.4 inches Wide Range).

Loss of Component Cooling Water Pumps 'B' and 'C'

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 CCW pumps 'B' and 'C' to trip.

B. The crew (ATC) will recognize the complete loss of CCW system, per 05-1-02-V-1, Loss of Component Cooling Water ONEP immediate action place the Mode switch to SHUT DOWN and trip Reactor Recirc Pumps.

Auto scram not available / Manual scram available

A. When the ATC places the Mode Switch to SHUTDOWN the mode switch will fail in STARTUP **(Event 8)**, no scram signal will be present.

B. The ATC will manually scram the reactor by arming and depressing the 4 RPS Manual Scram Pushbuttons per EN-OP-115.

C. The Crew will enter Scram and Turbine and Generator Trips ONEPs

Loss of All Offsite Power with bus 15AA lockout / Small LOCA

A. Two minutes after the mode switch is taken to shutdown all offsite power will be lost **(Event 9)**.

B. The Crew will recognize bus 15AA will be locked out and remain de-energized.

C. The CRS will enter EP-2, and call for RCIC to be started to maintain level.

D. A small Recirc loop leak will occur.

Division 2 EDG fails to Auto Energize ESF Bus 16AB.

A. The BOP will recognize Failure of Division 2 EDG to auto energize bus 16AB **(Event 10)**.

B. The BOP will recognize Division 2 EDG is running and manually close the EDG feeder breaker to 16AB 152-1608 from P864.

C. The BOP will verify the EDG powers the bus and recognize that Feeder breaker to 16BB3 is not closed by Green light on handswitch and alarm on P864-2A-E3

D. The BOP will verify SSW B is supplying cooling water.

E. Dispatch operator and/or electrical to investigate loss of 16BB3.

F. 16BB3 feeds 16B31 MCC that supplies power to RHR ‘B’ valves required for injection.

G. Recognizes that RHR ‘C’ pump trip.

H. Dispatch operator and/or electrical to investigate RHR pump ‘C’ trip.

I. After 5 minutes or when water level reaches -100 inches notify CRS that problem with 16BB3 has been found and waiting for control room to re-energize also RHR ‘C’ pump problem has been corrected and ready for restart.

Termination:

A. Once RPV is restored within 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	∗ ***Open P53-F001 (INSTR AIR SPLY HDR TO CTMT) prior to receiving alarm P680-4A2-E4 (CONT ROD DRIFT) and two or more control rods drifting.***	**When control rods to drift this could cause uneven flux distribution throughout the core which in turn could have detrimental effects reducing the margin of reactor safety limits.**
2	∗ **When Reactor Mode Switch fails, manually insert a scram by depressing the RPS MANUAL SCRAM PUSHBUTTONS or Manual initiate ATWS/ARI prior to installing Attachment 21, "De-energize scram solenoids."**	**Ensuring a reactor scram prevents entry into EP-2A (ATWS) procedure that would cause the crew to take compensatory actions that would complicate the event mitigation strategy.**
3	∗ **Restore 16AB, RHR B, and/or C injection prior to level reaching TAF (-167" Fuel Zone).**	**Maintaining adequate core cooling.**
*	**Critical Task** (As defined in NUREG 1021 Appendix D)	