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

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	**Total**	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	4	N/A			3	4	N/A		3	**20**			
	2	1	2	1				1	1			1	**7**			
	Tier Totals	**4**	**5**	**5**				**4**	**5**			**4**	**27**			
2. Plant Systems	1	2	3	3	3	3	2	1	2	2	3	2	**26**			
	2	2	1	1	1	1	1	1	1	1	1	1	**12**			
	Tier Totals	**4**	**4**	**4**	**4**	**4**	**3**	**2**	**3**	**3**	**4**	**3**	**38**			
3. Generic Knowledge and Abilities Categories		1		2		3		4					**10**	1	2	3	4
		3		**3**		**2**		**2**								

Note: 1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it is replaced by a K/A from another Tier 3 category.)

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

G* Generic K/As

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

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

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

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

ES-401							BWR Examination Outline Emergency and Abnormal Plant Evolutions - **Tier 1/Group 1 (RO)**	Form ES-401-1	
E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A2	G*	K/A Topic(s)	IR	#
295031 Reactor Low Water Level / 2					X		Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : EA2.04 Adequate core cooling	4.6*	17
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1			X				Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : EK3.01 Recirculation pump trip/runback: Plant-Specific	4.1	18
295038 High Off-site Release Rate / 9							NOT SAMPLED		--
600000 Plant Fire On Site / 8	X						Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: AK1.02 Fire Fighting	2.9	19
700000 Generator Voltage and Electric Grid Disturbances / 6			X				Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: AK3.02 Actions contained in abnormal operating procedure for voltage and grid disturbances	3.6	20
K/A Category Totals:	**3**	**3**	**4**	**3**	**4**	**3**	**Group Point Total:**		**20**

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO)	Form ES-401-1

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3	X						Knowledge of the operational implications of the following concepts as they apply to LOSS OF MAIN CONDENSER VACUUM: AK1.03 Loss of heat sink	3.6	21
295007 High Reactor Pressure / 3							NOT SAMPLED		
295008 High Reactor Water Level / 2							NOT SAMPLED		
295009 Low Reactor Water Level / 2		X					Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: AK2.02 Reactor water level control	3.9	22
295010 High Drywell Pressure / 5							NOT SAMPLED		
295011 High Containment Temp / 5			X				Knowledge of the reasons for the following responses as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY): AK3.01 Increased containment cooling: Mark-III	3.6	23
295012 High Drywell Temperature / 5							NOT SAMPLED		
295013 High Suppression Pool Temp. / 5							NOT SAMPLED		
295014 Inadvertent Reactivity Addition / 1				X			Ability to operate and/or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: AA1.06 Reactor/turbine pressure regulating system	3.3	24
295015 Incomplete SCRAM / 1							NOT SAMPLED		
295017 High Off-site Release Rate / 9							NOT SAMPLED		
295020 Inadvertent Cont. Isolation / 5 & 7					X		Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: AA2.03 Reactor power	3.7	25
295022 Loss of CRD Pumps / 1							NOT SAMPLED		
295029 High Suppression Pool Wtr Lvl / 5						X	2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	4.5	26
295032 High Secondary Containment Area Temperature / 5							NOT SAMPLED		
295033 High Secondary Containment Area Radiation Levels / 9		X					Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following: EK2.01 Area radiation monitoring system	3.8	27
295034 Secondary Containment Ventilation High Radiation / 9							NOT SAMPLED		
295035 Secondary Containment High Differential Pressure / 5							NOT SAMPLED		
295036 Secondary Containment High Sump/Area Water Level / 5							NOT SAMPLED		
500000 High CTMT Hydrogen Conc. / 5							NOT SAMPLED		
K/A Category Point Totals:	**1**	**2**	**1**	**1**	**1**	**1**	**Group Point Total:**		**7**

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

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode				X							X	Knowledge of RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: K4.07 Emergency generator load sequencing --------------------------------- 2.4.31 Knowledge of annunciator alarms, indications, or response procedures.	3.7 --- 4.2	28 --- 29
205000 Shutdown Cooling										X		Ability to manually operate and/or monitor in the control room: A4.01 SDC/RHR pumps	3.7	30
206000 HPCI												N/A for GGNS		
20700 Isol Condenser												N/A for GGNS		
209001 LPCS					X				X			Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM : K5.01 Indications of pump cavitation ----------------------------------- Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: A3.06 Lights and alarms	2.6 --- 3.6	31 --- 32
209002 HPCS								X				Ability to (a) predict the impacts of the following on the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.13 Low condensate storage tank level BWR-5,6	3.4	33
211000 SLC	X						X					Knowledge of the physical connections and/or cause-effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: K1.09 Core spray system: Plant-Specific ------------------------------------- Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: A1.04 Valve operations	3.2* --- 3.6	34 --- 35
212000 RPS						X						Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR PROTECTION SYSTEM : K6.01 A.C. electrical distribution	3.6	36
215003 IRM					X							Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : K5.03 Changing detector position	3.0	37
215004 Source Range Monitor				X								Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: K4.01 Rod withdrawal blocks	3.7	38

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System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
215005 APRM / LPRM			X									Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: K3.05 Reactor power indication	3.8	39
217000 RCIC		X										Knowledge of electrical power supplies to the following: K2.02 RCIC initiation signals (logic)	2.8*	40
218000 ADS	X											Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: K1.05 Remote shutdown system: Plant-Specific	3.9	41
223002 PCIS/Nuclear Steam Supply Shutoff			X									Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: K3.10 Reactor water cleanup	2.9	42
239002 SRVs		X										Knowledge of electrical power supplies to the following: K2.01 SRV solenoids	2.8*	43
259002 Reactor Water Level Control				X								Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: K4.12 Manual and automatic control of the system	3.5	44
261000 SGTS						X						Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : K6.04 Process radiation monitoring	2.9	45
262001 AC Electrical Distribution			X		X							Knowledge of the effect that a loss or malfunction of the A.C. ELECTRICAL DISTRIBUTION will have on following: K3.02 Emergency generators -- Knowledge of the operational implications of the following concepts as they apply to A.C. ELECTRICAL DISTRIBUTION: K5.01 Principle involved with paralleling two A.C. sources	3.8 ----- 3.1	46 --- 47
262002 UPS (AC/DC)										X		Ability to manually operate and/or monitor in the control room: A4.01 Transfer from alternative source to preferred source	2.8	48
263000 DC Electrical Distribution									X			Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: A3.01 Meters, dials, recorders, alarms, and indicating lights	3.2	49

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
264000 EDGs								X		X		Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.03 Operating unloaded, lightly loaded, and highly loaded ---------------------------------- Ability to manually operate and/or monitor in the control room: A4.05 Transfer of emergency generator (with load) to grid	3.4 --- 3.6	50 --- 51
300000 Instrument Air		X										Knowledge of electrical power supplies to the following: K2.01 Instrument air compressor	2.8	52
400000 Component Cooling Water											X	2.1.28 Knowledge of the purpose and function of major system components and controls.	4.1	53
K/A Category Point Totals:	**2**	**3**	**3**	**3**	**3**	**2**	**1**	**2**	**2**	**3**	**2**	**Group Point Total:**		**26**

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G*	K/A Topic(s)	IR	#
201001 CRD Hydraulic												NOT SAMPLED		
201002 RMCS												N/A for GGNS		
201003 Control Rod and Drive Mechanism												NOT SAMPLED		
201004 RSCS												N/A for GGNS		
201005 RCIS								X				Ability to (a) predict the impacts of the following on the ROD CONTROL AND INFORMATION SYSTEM (RCIS) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.04 Withdraw block: BWR-6	3.2	54
201006 RWM												N/A for GGNS		
202001 Recirculation	X											Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION SYSTEM and the following: K1.19 Feedwater flow	3.2	55
202002 Recirculation Flow Control												NOT SAMPLED		
204000 RWCU			X									Knowledge of the effect that a loss or malfunction of the REACTOR WATER CLEANUP SYSTEM will have on following: K3.06 Area radiation levels	2.6	56
214000 RPIS												N/A for GGNS		
215001 Traversing In-Core Probe												NOT SAMPLED		
215002 RBM												N/A for GGNS		
216000 Nuclear Boiler Inst.												NOT SAMPLED		
219000 RHR/LPCI: Torus/Pool Cooling Mode					X							Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: TORUS / SUPPRESSION POOL COOLING MODE: K5.04 Heat exchanger operation	2.9	57
223001 Primary CTMT and Aux.												NOT SAMPLED		
226001 RHR/LPCI: CTMT Spray Mode							X					Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: A1.10 Emergency generator loading	3.0	58
230000 RHR/LPCI: Torus/Pool Spray Mode												N/A for GGNS		
233000 Fuel Pool Cooling/Cleanup												NOT SAMPLED		
234000 Fuel Handling Equipment										X		Ability to manually operate and/or monitor in the control room: A4.01 Neutron monitoring system	3.7	59
239001 Main and Reheat Steam												NOT SAMPLED		
239003 MSIV Leakage Control												NOT SAMPLED		

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

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

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

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

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1. Emergency & Abnormal Plant Evolutions	1													4	3	7
	2													2	1	3
	Tier Totals													6	4	10
2. Plant Systems	1													3	2	5
	2													1 / 1	1	3
	Tier Totals													5	3	8

3. Generic Knowledge and Abilities Categories	1	2	3	4	Total	SRO 1	SRO 2	SRO 3	SRO 4	SRO Total
						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 outline sections (i.e., except for one category in Tier 3 of the SRO-only section, the "Tier Totals" in each K/A category shall not be less than two). (One Tier 3 radiation control K/A is allowed if it is replaced by a K/A from another Tier 3 category.)
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9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

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

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4									
295003 Partial or Complete Loss of AC / 6					X		Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: AA2.04 System lineups	3.7	76
295004 Partial or Total Loss of DC Pwr / 6									
295005 Main Turbine Generator Trip / 3					X		Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : AA2.04 Reactor pressure	3.8	77
295006 SCRAM / 1						X	2.2.25, Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.	4.2	78
295016 Control Room Abandonment / 7									
295018 Partial or Total Loss of CCW / 8									
295019 Partial or Total Loss of Inst. Air / 8									
295021 Loss of Shutdown Cooling / 4					X		Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING AA2.03 Reactor water level	3.5	79
295023 Refueling Acc / 8						X	2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	4.5	80
295024 High Drywell Pressure / 5									
295025 High Reactor Pressure / 3									
295026 Suppression Pool High Water Temp. / 5									
295027 High Containment Temperature / 5					X		Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III CONTAINMENT ONLY) EA2.01 Containment temperature: Mark-III	3.7	81
295028 High Drywell Temperature / 5									
295030 Low Suppression Pool Wtr Lvl / 5									
295031 Reactor Low Water Level / 2									
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1									
295038 High Off-site Release Rate / 9									
600000 Plant Fire On Site / 8						X	2.4.30 Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.	4.1	82
700000 Generator Voltage and Electric Grid Disturbances / 6									
K/A Category Totals:	**0**	**0**	**0**	**0**	**4**	**3**	**Group Point Total:**		**7**

ES-401	BWR Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (SRO)	Form ES-401-1

E/APE # / Name / Safety Function	K 1	K 2	K 3	A 1	A 2	G*	K/A Topic(s)	IR	#
295002 Loss of Main Condenser Vac / 3									
295007 High Reactor Pressure / 3					X		Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE AA2.02 Reactor power	4.1	83
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2									
295010 High Drywell Pressure / 5									
295011 High Containment Temp / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Temp. / 5									
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1					X		Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM AA2.02 Control rod position	4.2	84
295017 High Off-site Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1									
295029 High Suppression Pool Wtr Lvl / 5									
295032 High Secondary Containment Area Temperature / 5									
295033 High Secondary Containment Area Radiation Levels / 9									
295034 Secondary Containment Ventilation High Radiation / 9									
295035 Secondary Containment High Differential Pressure / 5									
295036 Secondary Containment High Sump/Area Water Level / 5									
500000 High CTMT Hydrogen Conc. / 5						X	2.4.41 Knowledge of the emergency action level thresholds and classifications.	4.6	85
K/A Category Point Totals:	**0**	**0**	**0**	**0**	**2**	**1**	**Group Point Total:**		**3**

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

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
203000 RHR/LPCI: Injection Mode											X	G2.1.23: Ability to perform specific system and integrated plant procedures during all modes of plant operations	4.4	86
205000 Shutdown Cooling														
206000 HPCI												N/A for GGNS		
20700 Isol Condenser												N/A for GGNS		
209001 LPCS														
209002 HPCS														
211000 SLC														
212000 RPS								X				Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.11 Main steamline isolation valve closure	4.1	87
215003 IRM														
215004 Source Range Monitor														
215005 APRM / LPRM														
217000 RCIC														
218000 ADS								X				Ability to (a) predict the impacts of the following on the AUTOMATIC DEPRESSURIZATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.05 Loss of A.C. or D.C. power to ADS valves	3.6	88
223002 PCIS/Nuclear Steam Supply Shutoff														
239002 SRVs											X	G2.2.12 Knowledge of Surveillance procedures	4.1	89
259002 Reactor Water Level Control														
261000 SGTS														
262001 AC Electrical Distribution														
262002 UPS (AC/DC)														
263000 DC Electrical Distribution														
264000 EDGs								X				Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL/JET) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.09 Loss of A.C. power	4.1	90
300000 Instrument Air														
400000 Component Cooling Water														
K/A Category Point Totals:	**0**	**0**	**0**	**0**	**0**	**0**	**0**	**3**	**0**	**0**	**2**	**Group Point Total:**		**5**

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

System # / Name	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A2	A 3	A 4	G*	K/A Topic(s)	IR	#
201001 CRD Hydraulic														
201002 RMCS												N/A for GGNS		
201003 Control Rod and Drive Mechanism														
201004 RSCS												N/A for GGNS		
201005 RCIS														
201006 RWM												N/A for GGNS		
202001 Recirculation								X				Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A2.12 Loss of reactor feedwater	3.8	91
202002 Recirculation Flow Control														
204000 RWCU														
214000 RPIS												N/A for GGNS		
215001 Traversing In-Core Probe														
215002 RBM												N/A for GGNS		
216000 Nuclear Boiler Inst.														
219000 RHR/LPCI: Torus/Pool Cooling Mode														
223001 Primary CTMT and Aux.														
226001 RHR/LPCI: CTMT Spray Mode														
230000 RHR/LPCI: Torus/Pool Spray Mode												N/A for GGNS		
233000 Fuel Pool Cooling/Cleanup														
234000 Fuel Handling Equipment				X								Knowledge of FUEL HANDLING EQUIPMENT design feature(s) and/or interlocks which provide for the following: K4.02 †Prevention of control rod movement during core alterations	4.1	92
239001 Main and Reheat Steam														
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator														
245000 Main Turbine Gen. / Aux.														
256000 Reactor Condensate														
259001 Reactor Feedwater											X	2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	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														
204000 RWCU														
K/A Category Point Totals:	**0**	**0**	**0**	**1**	**0**	**0**	**0**	**1**	**0**	**0**	**1**	**Group Point Total:**		**3**

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

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

Tier / Group (**Original**)	Randomly Selected K/A (**New**)	Reason for Rejection
RO T1/G1 295016 AA2.01	295016 AA2.02	Original KA: AA2. Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: AA2.01 Reactor power At GGNS we are unable to determine Reactor Power at the Remote Shutdown Panel or external from the main control room. The ability to determine or interpret reactor power for a control room abandonment would infer ATWS which is outside our design bases. Randomly selected New K/A – 295016, AA2.02, Reactor water level. Page 1 point totals not affected by this change. (Rev 1)
RO T1/G1 295027 2.1.19	295027 2.1.25	Original KA: 2.1.19 Ability to use plant computers to evaluate system or component status. At GGNS the use of plant computers to evaluate Containment Temperatures is limited due to the computer indication is an average of several instruments. Individual instrument indications are on the main Control Room panels. These indications are used more readily by ROs to determine the validity of RPV water level instrumentation by using Caution 1 of EOPs. Caution 1 uses a table and specific containment temperature instruments to determine RPV water level instrumentation validity. Randomly selected New K/A – 2.1.25, Ability to interpret reference materials, such as graphs, curves, tables, etc. Page 1 point totals not affected by this change. (Rev 1)
RO T1/G1 295037 EK3.08	295037 EK3.01	Original KA: EK3. Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : EK3.08 ATWS circuitry: Plant-Specific GGNS doesn't have an "ATWS Circuitry". Randomly selected New K/A – EK3.01, Recirculation pump trip/runback. Page 1 point totals not affected by this change. (Rev 1)

Tier / Group (Original)	Randomly Selected K/A (New)	Reason for Rejection
RO T2/G1 215004 K4.02	215004 K4.01	Original K/A: K4. Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: K4.02 Reactor SCRAM signals The SRMs do not provide Scram signals unless the shorting links are removed. There are no current procedures that will allow the shorting links to be removed at GGNS. Randomly selected K/A – K4.01, Withdrawal Blocks Page 1 point totals not affected by this change. (Rev 1)
RO T2/G1 223002 K3.20	223002 K3.10	Original K/A: K3. Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: K3.20 Standby gas treatment system At GGNS the Primary Containment Isolation system has no affect on the Standby gas treatment system. Randomly selected K/A – K3.10, RWCU Page 1 point totals not affected by this change. (Rev 1)
RO T2/G1 262002 A1.02	262002 A4.01	Original K/A: A1. Ability to predict and/or monitor changes in parameters associated with operating the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) controls including: A1.02 Motor generator outputs Motor Generators (MGs) are not used for Uninterruptable power at GGNS. Randomly selected K/A – A4.01 A4. Ability to manually operate and/or monitor in the control room: A4.01 Transfer from alternative source to preferred source. A1 had only one other selection that has a 2.4 importance factor. Operation's Representative reviewed the other KA and determined that the importance factor is below a 2.5, therefore A4 was randomly selected. Page 1 point totals were affected - A1 decreased by 1 in T2G1 and A4 increased by 1 in T2G1. (Rev 1)

Tier / Group (**Original**)	Randomly Selected K/A (**New**)	Reason for Rejection
RO T2/G2 234000 A4.02	234000 A4.01	Original K/A: A4. Ability to manually operate and/or monitor in the control room: A4.02 Control rod drive system At GGNS the CRD system is considered to be the hydraulic and mechanism portion. The Control Rod blade is part of the Reactor Vessel Internal system, therefore, there is no tie between Fuel Handling Equipment and the CRD system at GGNS. Randomly selected K/A – A4.01, Neutron Monitoring system.
SRO T1/G1 295006 2.2.36	295006 2.2.39	Original K/A: 2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. Unable to tie SRO only direction with the parent KA of SCRAM. Randomly selected K/A – 2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems.
SRO T1/G1 295023 2.1.19	295023 2.4.2	Original K/A: 2.4.3 Ability to identify post-accident instrumentation. Unable to tie SRO only direction with the parent KA of Refueling Accidents. There is no direct connection with post-accident instrumentation and a refuel accident. Randomly selected K/A – 2.4.2, Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.
SRO T1/G1 295006 2.2.39	295006 2.2.25	Original K/A: 2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems. This K/A is RO knowledge, discussed with Chief during review. Randomly selected K/A – 2.2.25, Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Tier / Group (**Original**)	Randomly Selected K/A (**New**)	Reason for Rejection
SRO T2/G2 295001 2.4.9	295001 2.4.8	Original K/A: 2.4.9 Knowledge of low power/shutdown implications in accident (e.g. loss of cooling coolant accident or loss of residual heat removal) mitigation strategies. This K/A doesn't fit the mitigation strategies for GGNS. GGNS uses the same EOPs and Abnormal procedures regardless of power level. Randomly selected K/A – 2.4.8, Knowledge of how abnormal operating procedures are used in conjunction with EOPs.

Facility: **Grand Gulf Nuclear Station**		Date of Examination: **12/04/2017**
Examination Level: **RO** ☒ SRO ☐		Operating Test Number: **LOT 12-2017**
Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R; D	**AR1 - Determine Primary Containment Water Level** (GJPM-OPS-2017IAR1) K/A 2.1.25: (3.9); 2.1.20: (4.6); 2.4.21: (4.0)
Conduct of Operations	S; M	**AR2 - Perform AC Lineup Surveillance** (GJPM-OPS-2017IAR2) K/A 2.1.31: (4.6); 2.2.12: (3.7); 2.1.20: (4.6)
Equipment Control	R; D	**AR3 - Determine Tagging Requirements** (GJPM-OPS-2017IAR3) K/A 2.2.41: (3.5); 2.2.13: (4.1)
Radiation Control		
Emergency Plan	R; N	**AR4 - Perform Emergency Notifications** (GJPM-OPS-2017IAR4) K/A 2.4.43: (3.2); 2.4.39: (3.9)

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

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

Facility: **Grand Gulf Nuclear Station**		Date of Examination: **12/04/2017**
Examination Level: RO ☐	**SRO** ☒	Operating Test Number: **LOT 12-2017**

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	R; M	**AS1 - Perform EOOS Risk Assessment** (GJPM-OPS-2017IAS1) K/A 2.1.39: (4.3)
Conduct of Operations	R; N	**AS2 - Review Completed Surveillance** (GJPM-OPS-2017IAS2) K/A 2.1.2: (4.4); 2.1.7: (4.7); 2.2.12: (4.1); 2.2.22: (4.7)
Equipment Control	R; D; P	**AS3 - Determine Impact on Plant Operations for Failed Relay** (GJPM-OPS-2017IAS3) K/A 2.2.41: (3.9); 2.2.22: (4.7); 2.2.36: (4.2)
Radiation Control	R; M	**AS4 - Authorize Emergency Exposure** (GJPM-OPS-2017IAS4) K/A 2.3.4: (3.7)
Emergency Plan	R; N	**AS5 - Perform Emergency Classification** (GJPM-OPS-2017IAS5) K/A 2.4.41: (4.6)

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

* Type Codes & Criteria:		
	(**C**)ontrol room, (**S**)imulator, or Class(**R**)oom	**(5)**
	(**D**)irect from bank (≤ 3 for ROs; ≤ 4 for SROs & RO retakes)	**(1)**
	(**N**)ew or (**M**)odified from bank (≥ 1)	**(4)**
	(**P**)revious 2 exams (≤ 1; randomly selected)	**(1)**

Facility: **Grand Gulf Nuclear Station**	Date of Examination: **12/04/2017**
Exam Level: **RO** ☒ SRO-I ☐ SRO-U ☐	Operating Test No.: **LOT 12-2017**

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

System / JPM Title	Type Code*	Safety Function
S1 - Manually Startup RCIC (GJPM-OPS-2017IS1) 217000 A4.04 (3.6)	A-N-S-L	2
S2 - Retest MSIV Slow Closure (GJPM-OPS-2017IS2) 239001 A2.11 (4.1)	P-A-D-S	3
S3 - Startup Shutdown Cooling (GJPM-OPS-2017IS3) 205000 A4.01 (3.7)	A-D-L-S	4
S4 - Start, Parallel and Load EDG (GJPM-OPS-2017IS4) 264000 A4.04 (3.7)	A-D-S	6
S5 - Startup H2 Recombiner (GJPM-OPS-2017IS5) 223001 A4.13 (3.4)	EN-D-S	5
S6 - Secure Standby Gas Treatment (GJPM-OPS-2017IS6) 261000 A4.02 (3.1)	EN-N-S	9
C1 - Bypass Control Rod in RACS (GJPM-OPS-2017ICR1) 201005 A2.04 (3.2)	D-C-L	7
S7 - Shift RR Pump B to Fast Speed (GJPM-OPS-2017IS7) 202001 A4.01 (3.7) **(RO ONLY)**	A-D-S	1
In-Plant Systems* (3 for RO); (3 for SRO-I); (3 or 2 for SRO-U)		
P1 - Align SP Cooling from RSP (GJPM-OPS-2017IP1) 219000 A4.01 (3.8)	A-E-D-L	5
P2 - Install N2 Bottles on ADS Air Supply (GJPM-OPS-2017IP2) 218000 A2.03 (3.4)	P-E-D-R-L	3
PB2 - Return Fire Water Pumps to Stby (GJPM-OPS-2017IPB2) 286000 4.05 (3.3)	D	8

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

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

Facility: **GRAND GULF NUCLEAR STATION** Date of Examination: **12/04/2017**

Exam Level: RO ☐ **SRO-I** ☒ SRO-U ☐ Operating Test No.: **LOT 12-2017**

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

System / JPM Title	Type Code*	Safety Function
S1 - Manually Startup RCIC (GJPM-OPS-2017IS1) 217000 A4.04 (3.6)	A-N-S-L	2
S2 - Retest MSIV Slow Closure (GJPM-OPS-2017IS2) 239001 A2.11 (4.3)	P-A-D-S	3
S3 - Startup Shutdown Cooling (GJPM-OPS-2017IS3) 205000 A4.01 (3.7)	A-D-L-S	4
S4 - Start, Parallel and Load EDG (GJPM-OPS-2017IS4) 264000 A4.04 (3.7)	A-D-S	6
S5 - Startup H2 Recombiner (GJPM-OPS-2017IS5) 223001 A4.13 (3.4)	EN-D-S	5
S6 - Secure Standby Gas Treatment (GJPM-OPS-2017IS6) 261000 A4.02 (3.1)	EN-N-S	9
CR1 - Bypass Control Rod in RACS (GJPM-OPS-2017ICR1) 201005 A2.04 (3.2)	D-C-L	7

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

System / JPM Title	Type Code*	Safety Function
P1 - Align SP Cooling from RSP (GJPM-OPS-2017IP1) 219000 A4.01 (3.7)	A-E-D-L	5
P2 - Install N2 Bottles on ADS Air Supply (GJPM-OPS-2017IP2) 218000 A2.03 (3.6)	E-D-R-L	3
PB2 - Return Fire Water Pumps to Stby (GJPM-OPS-2017IPB2) 286000 4.05 (3.3)	D	8

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

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

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

Examiners: ______________________________ Operators: ______________________________

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Event No.	Malf. No.	Event Type †	Event Description
1	N/A	N (BOP,CRS)	Transfer ESF Bus 17AC from ESF Transformer 21 to ESF Transformer 12
2	(r) fw211	C (ATC,CRS) TS (CRS)	Reactor Narrow Range Level C instrument oscillations
3	r21134h n41141c	TS (CRS) C (BOP,CRS) A (BOP,CRS)	ESF Transformer 12 Lockout with HPCS Diesel Generator auto start failures
4	z022021_24_53	C(ATC,CRS) A(CREW)	Control Rod 24-53FN drifting in
5	z022022_24_53	R (ATC) C(BOP,CRS) A(CREW) TS (CRS)	Control Rod 24-53FN stuck at position 32.
6	z021021_32_37	M(CREW)	Control Rod 32-27HJ drifting in. Reactor scram due to two controls drifting in
7	c11164	M(CREW)	Hydraulic Block ATWS > 5% RTP with SLC failure (EP-2, 2A) * **(CT-1) When control rods fail to scram, crew inserts all control rods to position 02 or beyond before exiting EP-2A** * **(CT-2) Inhibit ADS prior to automatic ADS valves opening during ATWS** * **(CT-3) During failure to scram conditions with power > 5% RTP, terminate and prevent all injection from all sources (except boron, CRD, and RCIC) as necessary to lower RPV level to below -70” wide range prior to exiting EP-2A**
8	fw123a(b)	C(BOP,CRS)	Reactor Feedwater Pump trip. * **(CT-4) Restores injection using Condensate/Feedwater to restore/maintain RPV level above -191” CFZ before exiting EP-2A**
9	c41263	C(ATC,CRS)	ESF Bus 15AA power loss

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

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

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

Reason for Revision: Editorial changes to enhance Technical Specification actions.

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

1. Transfer ESF Bus 17AC from ESF Transformer 21 to ESF Transformer 12.
2. Respond to Narrow Range C level transmitter oscillations.
3. Respond to an ESF Transformer 12 Lockout with failure of HPCS Diesel Generator to automatically start.
4. Respond to a control rod drifting in.
5. Respond to a stuck control rod.
6. Respond to a second control rod drifting in, resulting in a manual reactor scram.
7. Respond to a Hydraulic Block ATWS with power > 5% RTP.
8. Respond to a Reactor Feed Pump trip.
9. Respond to an ESF Bus 15AA power loss.

Initial Conditions: Plant is operating at 100% power.

Inoperable Equipment: None

Planned activities for this shift are:

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

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 80 minutes

Quantitative Attributes Table

Attribute	E3-304-1 Target	Actual	Description
Malfunctions after EOP entry	1-2	2	• Reactor Feed Pump trip (E8) • ESF Bus 15AA power loss (E9)
Abnormal Events	2-4	3	• ESF Transformer 12 lockout with a failure of HPCS Diesel Generator to automatically start (Loss of AC Power ONEP) (E3) • Control Rod 24-53FN drifting in (Control Rod/Drive Malfunctions ONEP) (E4) • Control Rod 24-53FN stuck at position 32 (Control Rod/Drive Malfunctions ONEP) (E5)
Major Transients	1-2	2	• Control Rod 32-37HJ drifting in – Reactor Scram (E6) • Hydraulic block ATWS with power > 5% RTP- SLC failure (E7)
EOP entries requiring substantive action	1-2	2	• EP-2 • EP-3
EOP contingencies requiring substantive action	0-2	1	• EP-2A
EOP based Critical Tasks	2-3	4	• **(CT-1) When control rods fail to scram, crew inserts all control rods to position 02 or beyond before exiting EP-2A** • **(CT-2) Inhibit ADS prior to automatic ADS valve opening during ATWS** • **(CT-3) During failure to scram conditions with power > 5% RTP, terminate and prevent all injection from all sources (except boron, CRD, and RCIC) as necessary to lower RPV level to below -70" wide range prior to exiting EP-2A** • **(CT-4) Restores injection using Condensate/Feedwater to restore/maintain RPV level above -191" CFZ before exiting EP-2A**
Normal Events	N/A	1	• Transfer ESF Bus 17AC from ESF Transformer 21 to ESF Transformer 12 (E1)
Reactivity Manipulations	N/A	1	• Lower core flow to 70 mlbm/hr using Reactor Recirc Flow Control Valves (E5)
Instrument / Component failures	N/A	6	• Narrow Range C level transmitter oscillations (E2) • ESF Transformer 12 lockout with a failure of HPCS Diesel Generator to automatically start (E3) • Control Rod 24-53FN drifting in (E4) • Control Rod 24-53FN stuck (E5) • Reactor Feed Water Pump A(B) trip (E8) • ESF Bus 15AA power loss (E9)
Total Malfunctions	N/A	8	• Narrow Range C level transmitter oscillations (E2) • ESF Transformer 12 lockout with a failure of HPCS Diesel Generator to automatically start (E3) • Control Rod 24-53FN drifting in (E4) • Control Rod 24-53FN stuck (E5) • Control Rod 32-37HJ drifting in - Reactor Scram(E6) • Hydraulic block ATWS with power > 5% RTP–SLC Failure (E7) • Reactor Feed Water Pump A(B) trip (E8) • ESF Bus 15AA power loss (E9)

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

RPS (Event 6)

ESF Power (Event 3)

Condensate (Event 8)

Failure to align alternate power to 4.16 KV or 6.9 KV buses (Event 3)

SCENARIO ACTIVITIES:

The plant is operating at 100% power.

Event 1 - Transfer ESF Bus 17AC from ESF Transformer 21 to ESF Transformer 12

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

Event 2 – Narrow Range C Level instrument oscillations (Triggered by Lead Examiner)

When ESF Bus 17AC has been transferred to ESF Transformer 12, Narrow Range C Level transmitter will begin oscillating. Crew will respond using ARI 04-1-02-1H13-P680-4A2-A2, RX WTR LVL SIG FAIL HI/LO, and manually select Narrow Range Level A or B. CRS will enter LCO TRM 6.3.7.

Event 3 - ESF Transformer 12 Lockout with HPCS Diesel Generator auto start failure (Triggered by Lead Examiner)

After Narrow Range Level A or B channel is selected and Tech Specs addressed, ESF Transformer 12 will lockout due to sudden pressure, causing a loss of power to ESF Bus 17AC. HPCS Diesel Generator will fail to auto start. BOP will recognize the failure of HPCS Diesel Generator to auto start and restore ESF Bus 17AC power from ESF Transformer 21 per 05-1-02-I-4, Loss of AC Power ONEP. CRS will enter TS 3.8.1.B for HPCS Diesel Generator inoperable.

Event 4 - Control Rod 24-53 drifting in (Triggered by Lead Examiner)

After ESF Bus 17AC power has been restored and Tech Specs addressed, Control Rod 24-53FN will begin drifting in. ATC will select Control Rod 24-53FN and apply a continuous insert signal per 05-1-02-IV-1, Control Rod/Drive Malfunctions ONEP.

Event 5 - Control Rod 24-53 stuck at position 32 (automatically triggered)

When Control Rod 24-53FN reaches position 32, it will become stuck. ATC will recognize and report Control Rod 24-53FN has stopped inserting. CRS will direct ATC to lower core flow to 70 mlbm/hr IAW Control Rod/Drive Malfunctions ONEP. ATC will lower core flow to 70 mlbm/hr using Recirc Flow Control Valves in fast detent. CRS will enter and direct actions from Reduction in Recirculation System Flow Rate ONEP, 05-1-02-III-3. After actions of Reduction in Recirculation System Flow Rate ONEP have been completed, CRS will direct actions IAW Control Rod/Drive Malfunctions ONEP for BOP to raise CRD drive water pressure in 25 psid increments and for ATC to attempt to insert Control Rod 24-53FN after each drive water pressure adjustment. When CRD Drive water pressure is raised to greater than 325 psid, **Event 6** will automatically be triggered.

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

Event 6 - Control Rod 32-27 drifting in (automatically triggered)

When CRD Drive Water pressure is raised above 325 psid, Control Rod 32-37HJ will begin to drift in. ATC will insert a manual scram per Control Rod/Drive Malfunctions ONEP. CRS will enter Reactor Scram ONEP, 05-1-02-I-1, and Turbine Generator Trip ONEP, 05-1-02-I-2.

NOTE: Event 6 can be triggered before CRD Drive Water pressure is raised to greater than 325 psid at the direction of the Lead Examiner.

Event 7 - Hydraulic Block ATWS > 5% RTP (No trigger required)

When reactor is scrammed, an ATWS occurs due to a hydraulic block of both scram discharge volumes. ATC will verify Reactor Recirc Pumps transfer to LFMGs, initiate ARI/RPT, inhibit ADS to prevent automatic operation **(CT-2)** and initiate and override HPCS IAW Reactor Scram ONEP, 05-1-02-I-1, immediate operator actions. CRS will enter EP-2A via EP-2. Reactor power will be above 5% RTP. ATC will initiate SLC which will fail to inject and initiate and override low pressure ECCS IAW Reactor Scram ONEP, 05-1-02-I-1, immediate operator actions. Terminate and Prevent of Feedwater is required because reactor power is above 5% RTP. RPV level is intentionally lowered below -70 inches wide range in order to lower core inlet subcooling and lower reactor power **(CT-3)**. Crew will install the necessary attachments to bypass RPS and RC&IS interlocks and insert controls rods via manual scrams and RC&IS **(CT-1)**. Suppression Pool Cooling will be maximized using RHR A and RHR B. Bypass valves will control reactor pressure during this event. Feedwater is available for RPV level control.

Event 8 - Reactor Feedwater Pump trip (Triggered by the Lead Examiner after reactor water level is stabilized below -70 inches wide range)

When reactor level lowers below -70 inches wide range, the in-service Reactor Feed Pump will trip. BOP will restore Feedwater injection to the RPV by starting the standby Reactor Feed Pump **(CT-4)** IAW 04-1-01-E12-1, Attachment 6, per 02-S-01-43, Transient Mitigation Strategy. An alternate success path would be CRS directing ATC to lower reactor pressure to 450 to 600 psig to allow RPV injection with Condensate Booster Pumps **(CT-4)** IAW 02-S-01-43, Transient Mitigation Strategy.

Event 9 – ESF Bus 15AA power loss (Triggered by Lead Examiner before controls rods are inserted)

After the running Reactor Feed Pump has tripped and RPV level has been stabilized, breaker 152-1514, ESF BUS 15AA FDR FM XFMR 11, will trip. Division 1 Diesel Generator will automatically restore power to ESF Bus 15AA. The ATC will recognize the loss of override function for LPCS and RHR A and override the associated pumps and injection valves IAW EP-2A and 02-S-01-43, Transient Mitigation Strategy. CRS will direct the ATC to restore Instrument Air to Containment IAW 05-1-02-I-4, Loss of AC Power ONEP. ATC will restore Instrument Air to Containment by opening P53-F001.

NOTE: While crew is responding to Event 9, report Attachments needed for scramming and driving control rods (18, 19, and 20) are complete to allow crew to prioritize actions.

After crew has begun inserting control rods, at direction of the Lead Examiner, the control rods are allowed to be fully inserted with the next scram. CRS transitions from EP-2A to EP-2 and RPV level restoration is directed.

NOTE: Examiner watching ATC should assist the Lead Examiner in determining when to allow all control rods to be inserted.

The exercise ends when controls rods are inserted, EP-2A has been exited and RPV water level band has been changed to between +11.4 inches and +53.5 inches narrow range.

Critical Task	(CT-1) When control rods fail to scram, crew inserts all control rods to position 02 or beyond before exiting EP-2A	(CT-2) Inhibit ADS prior to automatic ADS valve opening during ATWS
EVENT	7	7
Safety Significance	Failure to effect shutdown of the reactor when a RPS setpoint has been exceeded would unnecessarily extend the level of degradation of the safety of the plant. This could further degrade into damage to the principle fission product barriers if left unmitigated. The crew is authorized by Conduct of Operations to take mitigating actions when automatic safety systems fail to perform their intended function. Action to shut down the reactor is required when RPS and control rod drive systems fail IAW EP-2A.	Steps in EP-2A may intentionally lower RPV water level below the ADS setpoint to reduce reactor power. Permitting automatic ADS initiation may be undesirable for the following reasons: • ADS actuation can impose a severe thermal transient on the RPV and may complicate efforts to control RPV water level • If only RCIC is available for injection, ADS actuation may directly lead to loss of adequate core cooling and subsequent core damage • The conditions assumed in the design of the ADS actuation logic may not exist when the specified actions are being carried out • The operating crew can draw on much more information than available to ADS logic and can better judge, based on instructions contained in procedure, when and how to depressurize the RPV • Subsequent steps provide explicit and detailed instructions for RPV water level control and identify the specific conditions when RPV blowdown is required • Rapid, uncontrolled injection of relatively cold, unborated water could occur as RPV pressure decreases. If reactor is not shutdown or if shutdown margin is small, this could add sufficient positive reactivity to cause power excursion large enough to damage the core Automatic initiation of ADS is therefore inhibited upon entry of EP-2A.
Cueing	Manual scram is initiated and numerous control rods indicate beyond position 02. Reactor power indicating > 5% RTP on APRMs on panel P680. APRM downscale lights on panel P680 extinguished.	ADS Timer initiated alarm on P601
Performance Indicator	Operator selects control rod gangs by depressing the respective pushbuttons on panel P680 and inserts the rods by depressing the IN-TIMER SKIP pushbutton. Operator resets reactor scram signal with key-locked switches on panel P680 and inserts manual reactor scram using scram pushbuttons on panel P680.	Manipulation of ADS A and ADS B MANUAL INHIBIT switches on panel P601 vertical section.
Performance Feedback	Operator selecting and inserting control rods indicated by rod position decreasing to 00 for selected rods on panel P680. Control rod movement on subsequent reactor scrams. Reactor power lowering.	Inhibit switches click into INHIBIT position on panel P601 vertical section. White indicating light on ADS A and ADS B MANUAL INHIBIT switches illuminate. Receipt of ADS/SRV A and ADS/SRV B OOSVC alarms on panel P601/18A-H2 and P601/19A-H2.
Justification for the chosen performance limit	There is no time limit for effecting complete reactor shutdown via control rod insertion. For the timeframe of this scenario, containment limits are not challenged and power oscillations are not experienced. However, if the failure to scram EP were to be exited, other procedures would not provide the guidance necessary to achieve reactor shutdown. Before exiting EP-2A ensures guidance to effect reactor shutdown is not removed.	The 105 second ADS timer allows sufficient time for the crew to recognize and override automatic operation of the system. As long as ADS is inhibited before ADS valves open, reactor pressure will not be reduced to the shutoff heads of high volume, cold water systems.

BWR Owners Group Appendix	App. B, step RC/Q6, RC/Q-7	App. B, step RC/Q-6
Licensed Bases Documents	02-S-01-40, EP Technical Bases, Attachment V, Step Q-1 UFSAR Chapter 15.8	02-S-01-40, EP Technical Bases, Attachment V, Step 1 UFSAR Chapter 15.8

*** If an operator or the crew significantly deviates from, or fails to, follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review (NUREG 1021, Appendix D)**

Critical Task	(CT-3) During failure to scram conditions with power > 5%, terminate and prevent all injection from all sources (except boron, CRD, and RCIC) as necessary to lower RPV level to below -70" wide range prior to exiting EP-2A	(CT-4) Restores injection using Condensate/Feedwater to restore/maintain RPV level above -191" CFZ before exiting EP-2A
EVENT	7	8
Safety Significance	Regarding lowering level below -70" wide range, to prevent or mitigate the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal-hydraulic instabilities. RPV water level is lowered sufficiently below the elevation of the feedwater sparger nozzles. This places the feedwater spargers in the steam space providing effective heating of the relatively cold feedwater and eliminating the potential for high core inlet subcooling. For conditions that are susceptible to oscillations, the initiation and growth of oscillations is principally dependent upon the subcooling at the core inlet; the greater the subcooling, the more likely oscillations will commence and increase in magnitude. 24" below the lowest nozzle in the feedwater sparger has been selected as the upper bound of the RPV water level control band. This water level is sufficiently low that steam heating of the injected water will be at least 65% to 75% effective (i.e., the temperature of the injected water will be increased to 65% to 75% of its equilibrium value in the steam environment). This water level is sufficiently high that most plants without the capability to readily defeat the low RPV water level MSIV isolation should be able to control RPV water level with feedwater pumps to preclude the isolation.	If RPV water level cannot be restored and maintained above the Minimum Steam Cooling RPV Water Level (-191" CFZ), emergency RPV depressurization is performed to maximize injection flow. Emergency depressurization is undesirable under ATWS conditions since the core response is difficult to predict and the risk of power excursions is increased.
Cueing	Manual scram is initiated and numerous control rods indicate beyond position 02 and reactor power is > 5% on panel P680 indications and SPDS and RPV level is > -70" wide range on SPDS and PDS.	Reactor Feed Pump trip annunciators and Feedwater flow and RPV level lowering on indicators on panel P680 and PDS and SPDS.
Performance Indicator	Operator initiates HPCS using HPCS manual initiation switch, then secures HPCS pump and overrides the HPCS injection valve closed. Operator initiates Div 1 and Div 2 ECCS with their respective manual initiation switches and overrides the associated injection valves closed and secures LPCS and RHR C pumps. RHR A and B pumps may be left running for Suppression Pool Cooling. Operator manipulates Master Level Controller or Startup Level Controller in MANUAL and lowers output to 0%. Operator ensures N21-F009A and B and N21-F040 closed.	Operator manipulates switches on panel P680 panel to start the standby Reactor Feed Pump Alternately, operator lowers RPV pressure using Bypass Valves or SRVs to allow injection with Condensate Booster Pumps.
Performance Feedback	Feedwater flow indication on panel P680 and SPDS indicate zero. Valves N21-F009A and B and N21-F040 green lights illuminated. Master Level Controller output indicates 0% on panel P680. High Pressure Core Spray, Low Pressure Core Spray and RHR C pump and injection valve override annunciators illuminated on panel P601. RHR A and RHR B injection valve override annunciators illuminated on panel P601.	Feedwater flow and RPV level rising on panel P680 and PDS and SPDS.

Justification for the chosen performance limit	Applicability for this CT is during EP-2A conditions where it is necessary to lower level to control power with no high energy input into the primary containment. There is no time limit for this lowering level, but it establishes margin to conditions where fuel damaging power oscillations may theoretically occur. Before exiting EP-2A was chosen because other procedures would not provide the guidance necessary to establish margin for power oscillation mitigation. Before exiting EP-2A ensures guidance to effect this control is not removed. NOTE – This critical task must be evaluated carefully based on the level changes. If power is reduced significantly below 5%, reactor water level may continue to rise above -70" with only CRD and SLC while driving control rods. This would not result in an UNSAT on this critical task.	The Minimum Steam Cooling RPV Water Level (-191" CFZ) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Maintaining RPV water level above the Minimum Steam Cooling RPV Water Level thus ensures that the core remains adequately cooled.
BWR Owners Group Appendix	App. B, Contingency #5 Step C5-4	App. B, Contingency #5 Step C5-4
Licensed Bases Documents	02-S-01-40, EP Technical Bases, Attachment V, Step L-7 UFSAR Chapter 15.8	02-S-01-40, EP Technical Bases, Attachment V, Step L-9 UFSAR Chapter 15.8

*** If an operator or the crew significantly deviates from, or fails to, follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review (NUREG 1021, Appendix D)**

Simulator Setup:

A. Initialization

1. Log off all simulator PDS and SPDS computers (PDS and SPDS must come up after the simulator load for proper operation).

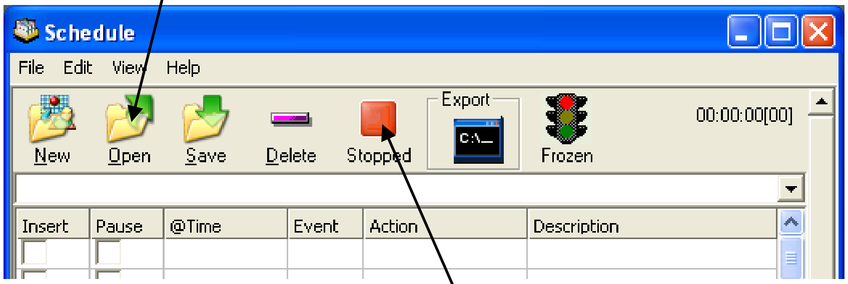
2. Startup the simulator using Simulator Instructor’s Job Aid section 7.3.

> *Note:*
> ***Prior to running the Schedule File, ensure no Event Files are Open.*** *If an existing Event File is Open prior to running the Schedule File, then any associated Event Files will not automatically load.*

3. Open Schedule.exe and Director.exe by clicking on the Icon in the Thunder Bar.

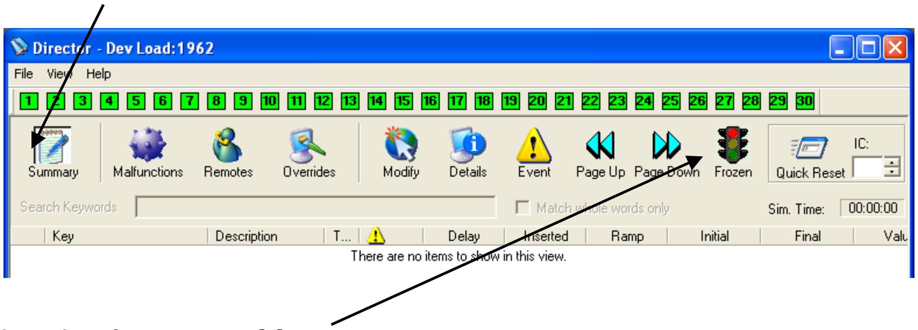
4. Set the Simulator to **IC-101** and perform switch check (Using Quick Reset in Director).

5. Click on “**Open”** in the Schedule window and Open Schedule File “**12-2017 NRC Exam Scenario 2.sch”** (in the Schedule Directory)



6. In Schedule window, click on the **“Stopped**” red block. The red block will change to a green arrow and indicate the scenario is active (“**Running**”).

7. Click the Summary tab in the Director window. Verify the schedule files are loaded and opened per Section B below. (Note: Any actions in the schedule file without a specific time will not load into the director until triggered.)



8. Take the simulator out of freeze.

9. Log on to all simulator PDS and SPDS computers.

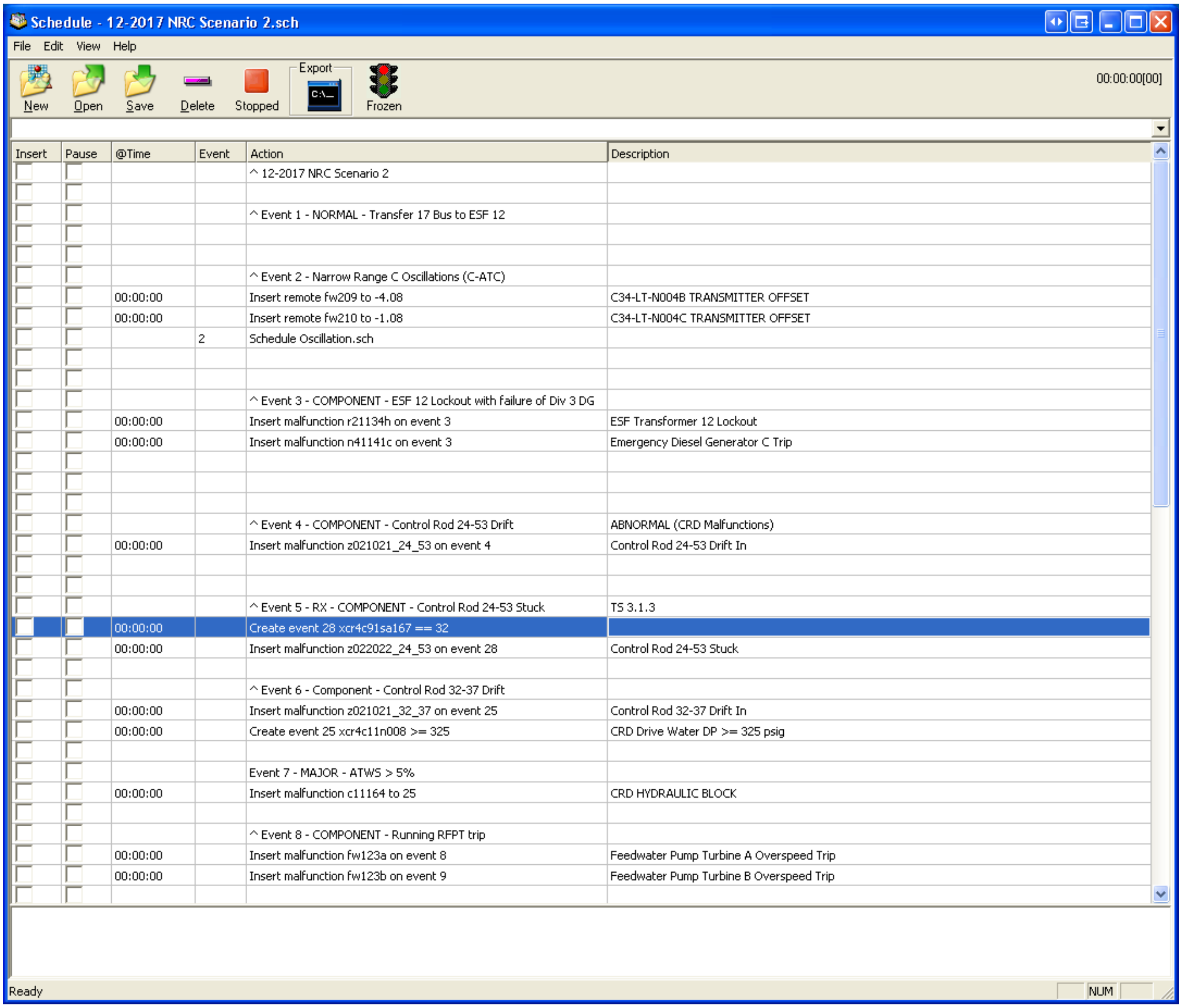
10. Verify or perform the following:

    - IC-101
    - Ensure all procedures are marked as indicated for turnover conditions
    - Advance all chart recorders and ensure all pens are inking properly
    - Clear any graphs and trends off PDS and SPDS

11. Run through any alarms and ensure alarms are on. **(Note: On T-Rex, to verify alarms are ON, the indicator will indicate "Alarms On").**

12. Place the simulator in Freeze.

File loaded verification:



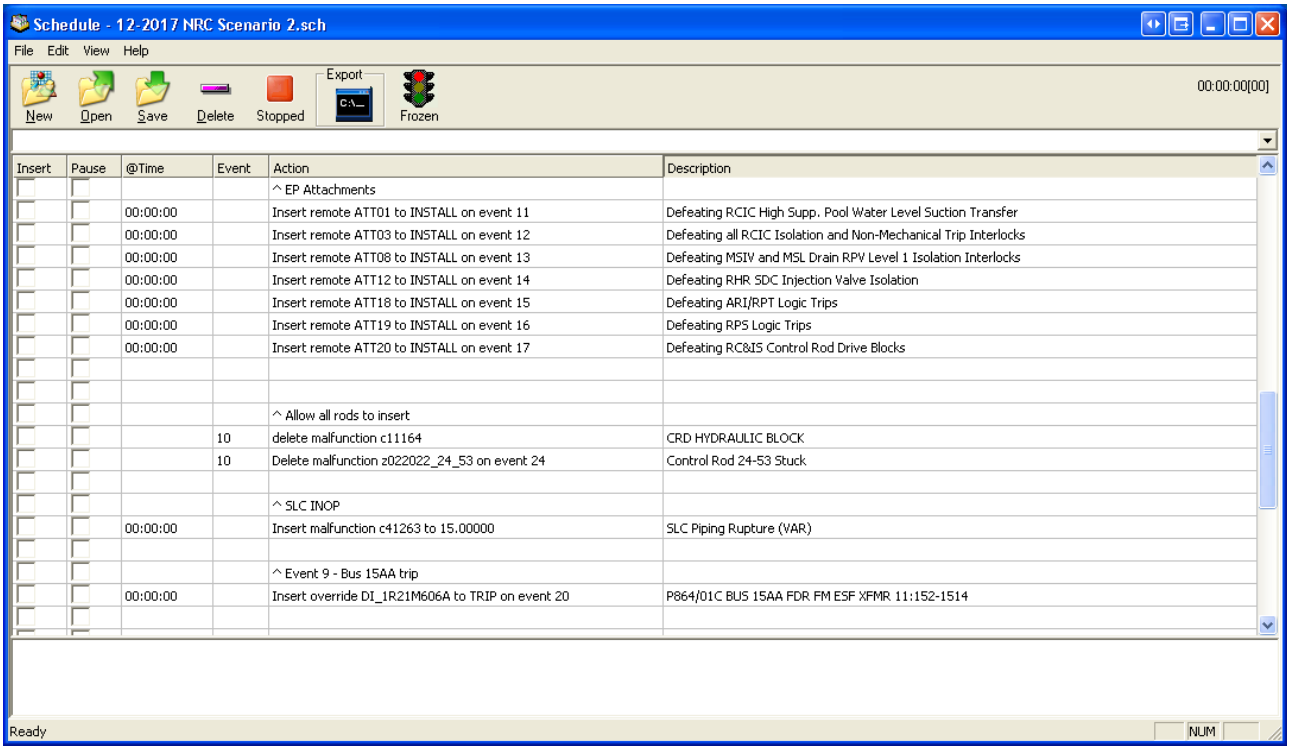
Schedule - 12-2017 NRC Scenario 2.sch

File Edit View Help

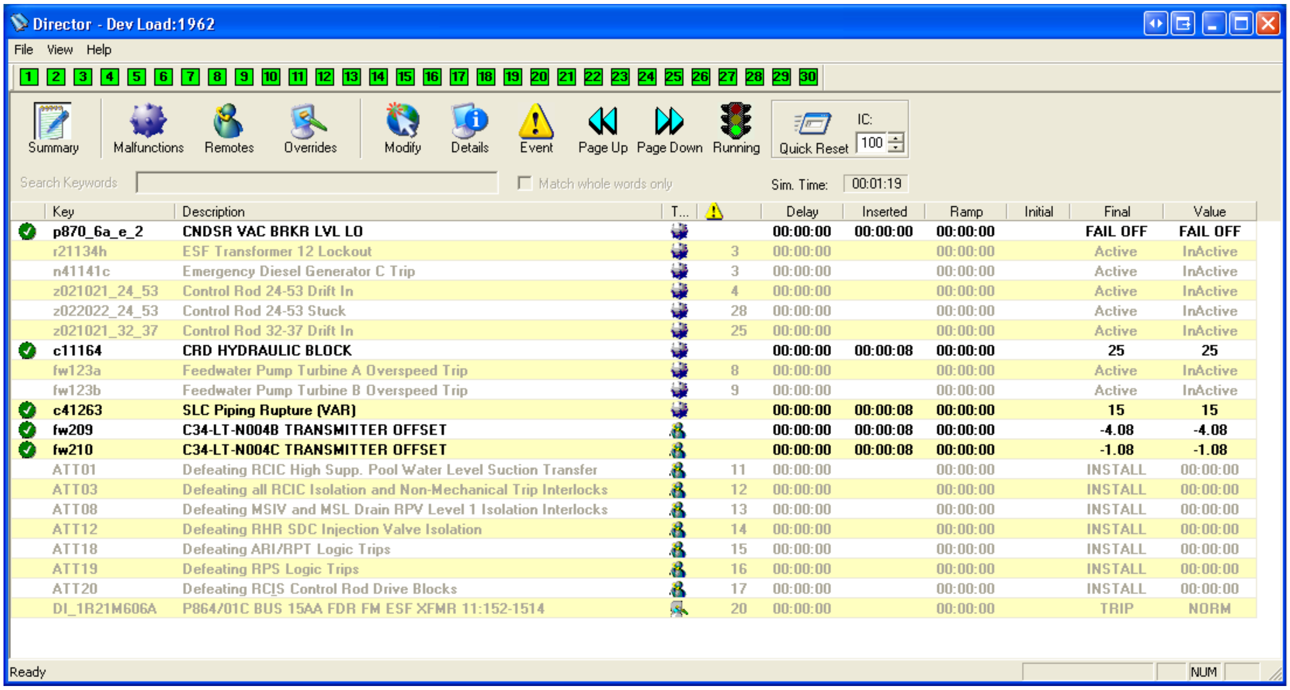
New Open Save Delete Stopped Export Frozen 00:00:00[00]

Insert	Pause	@Time	Event	Action	Description
				^ 12-2017 NRC Scenario 2	
				^ Event 1 - NORMAL - Transfer 17 Bus to ESF 12	
				^ Event 2 - Narrow Range C Oscillations (C-ATC)	
		00:00:00		Insert remote fw209 to -4.08	C34-LT-N004B TRANSMITTER OFFSET
		00:00:00		Insert remote fw210 to -1.08	C34-LT-N004C TRANSMITTER OFFSET
			2	Schedule Oscillation.sch	
				^ Event 3 - COMPONENT - ESF 12 Lockout with failure of Div 3 DG	
		00:00:00		Insert malfunction r21134h on event 3	ESF Transformer 12 Lockout
		00:00:00		Insert malfunction n41141c on event 3	Emergency Diesel Generator C Trip
				^ Event 4 - COMPONENT - Control Rod 24-53 Drift	ABNORMAL (CRD Malfunctions)
		00:00:00		Insert malfunction z021021_24_53 on event 4	Control Rod 24-53 Drift In
				^ Event 5 - RX - COMPONENT - Control Rod 24-53 Stuck	TS 3.1.3
		00:00:00		Create event 28 xcr4c91sa167 == 32	
		00:00:00		Insert malfunction z022022_24_53 on event 28	Control Rod 24-53 Stuck
				^ Event 6 - Component - Control Rod 32-37 Drift	
		00:00:00		Insert malfunction z021021_32_37 on event 25	Control Rod 32-37 Drift In
		00:00:00		Create event 25 xcr4c11n008 >= 325	CRD Drive Water DP >= 325 psig
				Event 7 - MAJOR - ATWS > 5%	
		00:00:00		Insert malfunction c11164 to 25	CRD HYDRAULIC BLOCK
				^ Event 8 - COMPONENT - Running RFPT trip	
		00:00:00		Insert malfunction fw123a on event 8	Feedwater Pump Turbine A Overspeed Trip
		00:00:00		Insert malfunction fw123b on event 9	Feedwater Pump Turbine B Overspeed Trip

Ready NUM



Insert	Pause	@Time	Event	Action	Description
				^ EP Attachments	
		00:00:00		Insert remote ATT01 to INSTALL on event 11	Defeating RCIC High Supp. Pool Water Level Suction Transfer
		00:00:00		Insert remote ATT03 to INSTALL on event 12	Defeating all RCIC Isolation and Non-Mechanical Trip Interlocks
		00:00:00		Insert remote ATT08 to INSTALL on event 13	Defeating MSIV and MSL Drain RPV Level 1 Isolation Interlocks
		00:00:00		Insert remote ATT12 to INSTALL on event 14	Defeating RHR SDC Injection Valve Isolation
		00:00:00		Insert remote ATT18 to INSTALL on event 15	Defeating ARI/RPT Logic Trips
		00:00:00		Insert remote ATT19 to INSTALL on event 16	Defeating RPS Logic Trips
		00:00:00		Insert remote ATT20 to INSTALL on event 17	Defeating RC&IS Control Rod Drive Blocks
				^ Allow all rods to insert	
			10	delete malfunction c11164	CRD HYDRAULIC BLOCK
			10	Delete malfunction z022022_24_53 on event 24	Control Rod 24-53 Stuck
				^ SLC INOP	
		00:00:00		Insert malfunction c41263 to 15.00000	SLC Piping Rupture (VAR)
				^ Event 9 - Bus 15AA trip	
		00:00:00		Insert override DI_1R21M606A to TRIP on event 20	P864/01C BUS 15AA FDR FM ESF XFMR 11:152-1514



Key	Description	T...	⚠	Delay	Inserted	Ramp	Initial	Final	Value
p870_6a_e_2	CNDSR VAC BRKR LVL LO			00:00:00	00:00:00	00:00:00		FAIL OFF	FAIL OFF
r21134h	ESF Transformer 12 Lockout		3	00:00:00		00:00:00		Active	InActive
n41141c	Emergency Diesel Generator C Trip		3	00:00:00		00:00:00		Active	InActive
z021021_24_53	Control Rod 24-53 Drift In		4	00:00:00		00:00:00		Active	InActive
z022022_24_53	Control Rod 24-53 Stuck		28	00:00:00		00:00:00		Active	InActive
z021021_32_37	Control Rod 32-37 Drift In		25	00:00:00		00:00:00		Active	InActive
c11164	CRD HYDRAULIC BLOCK			00:00:00	00:00:08	00:00:00		25	25
fw123a	Feedwater Pump Turbine A Overspeed Trip		8	00:00:00		00:00:00		Active	InActive
fw123b	Feedwater Pump Turbine B Overspeed Trip		9	00:00:00		00:00:00		Active	InActive
c41263	SLC Piping Rupture (VAR)			00:00:00	00:00:08	00:00:00		15	15
fw209	C34-LT-N004B TRANSMITTER OFFSET			00:00:00	00:00:08	00:00:00		-4.08	-4.08
fw210	C34-LT-N004C TRANSMITTER OFFSET			00:00:00	00:00:08	00:00:00		-1.08	-1.08
ATT01	Defeating RCIC High Supp. Pool Water Level Suction Transfer		11	00:00:00		00:00:00		INSTALL	00:00:00
ATT03	Defeating all RCIC Isolation and Non-Mechanical Trip Interlocks		12	00:00:00		00:00:00		INSTALL	00:00:00
ATT08	Defeating MSIV and MSL Drain RPV Level 1 Isolation Interlocks		13	00:00:00		00:00:00		INSTALL	00:00:00
ATT12	Defeating RHR SDC Injection Valve Isolation		14	00:00:00		00:00:00		INSTALL	00:00:00
ATT18	Defeating ARI/RPT Logic Trips		15	00:00:00		00:00:00		INSTALL	00:00:00
ATT19	Defeating RPS Logic Trips		16	00:00:00		00:00:00		INSTALL	00:00:00
ATT20	Defeating RCIS Control Rod Drive Blocks		17	00:00:00		00:00:00		INSTALL	00:00:00
DI_1R21M606A	P864/01C BUS 15AA FDR FM ESF XFMR 11:152-1514		20	00:00:00		00:00:00		TRIP	NORM

Procedures that may be used in this scenario:

Procedure No.	Rev	Procedure Title
04-1-01-C11-1	154	Control Rod Drive Hydraulic System
04-1-01-C41-1	123	Standby Liquid Control System
04-1-01-E12-1	147	Residual Heat Removal System
04-1-01-N21-1	74	Feedwater System
04-1-01-N32-2	33	Turbine Generator Control
04-1-01-R21-17	10	ESF Bus 17AC
04-1-02-1H13-P601	161	Alarm Response Instruction Panel No.: 1H13-P601
04-1-02-1H13-P680	233	Alarm Response Instruction Panel No.:1H13-P680
04-1-02-1H13-P864	31	Alarm Response Instruction Panel No.: 1H13-P864
04-1-02-1H13-P870	154	Alarm Response Instruction Panel No.: 1H13-P870
04-S-02-SH13-P807	32	Alarm Response Instruction Panel No.: SH13-P807
05-1-02-I-1	130	Reactor Scram
05-1-02-I-2	37	Turbine and Generator Trips
05-1-02-I-4	51	Loss of AC Power
05-1-02-III-3	115	Reduction In Recirculation System Flow Rate
05-1-02-III-5	49	Automatic Isolations
05-1-02-IV-1	117	Control Rod / Drive Malfunctions
05-S-01-EP-1	36	Emergency / Severe Accident Procedure Support Documents
05-S-01-EP-2	45	RPV Control
Tech Spec 3.1.3		
Tech Spec 3.8.1		
Tech Spec TR6.3.7		

Expected Alarms:

P601-16A-E1, 4.16 KV BUS 17AC INCM FDR 152-1705
P680-2A-C9, DFCS TROUBLE
P680-4A2-A2, RX WTR LVL SIG FAIL HI/LO
P680-3A-A3, RX LVL 40”/32” HI/LO
P807-4A-B3, ESF XFMR 12 LOCKOUT TRIP
P807-4A-F4, ESF XFMR 12 TROUBLE
P807-1A-B1, SWYD XFMR T3 INCM FDR 152-1905 TRIP
P807-1A-B2, ESF DIST BUSES INCM FDR 152-1903 TRIP
P807-1A-B3, ESF DIST BUSES INCM FDR 152-1904 TRIP
P601-16A-A1, HPCS GEN TRIP/LOCKOUT
P601-16A-A2, HPCS DSL ENG TRIP
P601-16A-D3, HPCS DSL ENG TROUBLE
P601-16A-C1, 4.16KV BUS 17AC INCM FDR 152-1704 TRIP
P601-16A-F2, HPCS SYS UNDERVOLT
P601-16A-G1, 480V MCC 17B01 UNDERVOLT
P601-16A-H1, HPCS SYS NOT READY FOR AUTO START
P601-16A-E4, HPCS JKY PMP DISCH PRESS LO
P870-9A-3B, SSW DIV 3 OOSVC
P870-9A-F1, DG 13 TRIP UNIT TROUBLE
P870-9A-F2, SSW LOOP C LEAK HI
P870-9A-G1, DG 13 FUEL OIL XFER PMP CONT PWR FAIL
P680-4A2-E4 CONT ROD DRIFT

INITIAL CONDITIONS

A. Plant Status: 100% power, middle of cycle

B. Tech. Spec. Limitations in effect: None

C. Significant problems/abnormalities: None

D. Integrated Risk: Green

E. Division Work Week: Division 3

F. Evolutions/maintenance for the up-coming shift :

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

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

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

Event No.	Malf. No.	Event Type †	Event Description
1	N/A	N (BOP,ATC,CRS)	Transfer RPS Bus B from normal to alternate power supply
2	p864_2a_d_2	TS (CRS) C (BOP,CRS)	Division 2 Diesel Generator lube oil leak
3	ltb21n091b ltb21n091f	I (ATC,BOP,CRS) A(CREW) TS (CRS)	Division 2 ECCS initiation on spurious RPV low level signal
4	fw163c	R (ATC,CRS) A(CREW)	Loss of condenser vacuum
5	r21135 rr063b	M(CREW)	LOP/LOCA (EP-2, 3)
6	e22052	C(ATC,BOP,CRS)	HPCS Pump trip *** (CT-1) Inhibit ADS prior to automatic ADS valve opening during a LOCA** *** (CT-2) When RPV level lowers to -160" wide range and cannot be maintained above -191" CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191" CFZ. (Momentary shrink below -191" due to automatic SRV closure does not constitute failure of this critical task)**
7	rr040a rr041a	C(ATC,CRS)	Failure of Division 1 ECCS to automatically initiate *** (CT-3) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually initiates ECCS systems for injection prior to RPV pressure lowering below 300 psig**

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

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

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

<u>Reason for Revision:</u> Editorial changes to enhance Technical Specification actions.

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

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

Initial Conditions: Plant is operating at 100% power.

Inoperable Equipment:

- TBCW Pump C is tagged out for motor oil replacement.
- CRD Pump B is tagged out of service for oil replacement in the speed increaser.

Turnover:

Planned activities for this shift are:

- Transfer RPS Bus B from normal to alternate power supply IAW SOI 04-1-01-C71-1, Section 5.1, in preparation for preventative maintenance on the RPS B Motor Generator.
- The Motor Generator will be tagged out on the next shift.
- No scram or isolation surveillances are in progress or planned for this shift.

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 75 minutes

Quantitative Attributes Table

Attribute	E3-304-1 Target	Actual	Description
Malfunctions after EOP entry	1-2	2	• HPCS Pump trip (E6) • Failure of Division 1 ECCS to automatically initiate (E7)
Abnormal Events	2-4	2	• Spurious Division 2 ECCS initiation (Loss of One or Both RPS Buses ONEP and Automatic Isolations ONEP) (E3) • Loss of condenser vacuum (Loss of Condenser Vacuum ONEP) (E4)
Major Transients	1-2	2	• LOP • LOCA
EOP entries requiring substantive action	1-2	2	• EP-2 • EP-3
EOP contingencies requiring substantive action	0-2	2	• EP-2 Alternate Level Control • EP-2 Emergency Depressurization
EOP based Critical Tasks	2-3	3	• **(CT-1) Inhibit ADS prior to automatic ADS valve opening during a LOCA** • **(CT-2) When RPV level lowers to -160" wide range and cannot be maintained above -191" CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191" CFZ. (Momentary shrink below -191" due to automatic SRV closure does not constitute failure of this critical task)** • **(CT-3) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually initiates ECCS systems for injection prior to RPV pressure lowering below 300 psig**
Normal Events	N/A	1	• Transfer RPS Bus B from normal to alternate power supply (E1)
Reactivity Manipulations	N/A	1	• Lower core flow to 70 mlbm using Reactor Recirc Flow Control Valves (E4)
Instrument / Component failures	N/A	5	• Division 2 Diesel Generator lube oil leak (E2) • Spurious Division 2 ECCS initiation (E3) • Loss of vacuum (E4) • HPCS Pump trip (E6) • Failure of Division 1 ECCS to automatically initiate (E7)
Total Malfunctions	N/A	6	• Division 2 Diesel Generator lube oil leak (E2) • Spurious Division 2 ECCS initiation (E3) • Loss of vacuum (E4) • LOP/LOCA (E5) • HPCS Pump trip (E6) • Failure of Division 1 ECCS to automatically initiate (E7)

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

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

SCENARIO ACTIVITIES:

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

Event 1 – Transfer RPS Bus B from normal to alternate power supply

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

Event 2 - Division 2 Diesel Generator lube oil leak (Triggered by Lead Examiner)

After RPS Bus B is transferred to alternate power, annunciator “DIV 2 DSL GEN TROUBLE” will alarm. BOP will dispatch plant operator to investigate. After 2 minutes, the plant operator will report lube oil spraying out from the Division 2 Lube Oil Circulating Pump discharge piping and lube oil sump level is 20” below the top of the sump, which is less than 350 gallons. The BOP will place Division 2 Diesel Generator in the MAINTENANCE Mode IAW SOI 04-1-01-P75-1, Standby Diesel Generator System, Attachment VI. The CRS will enter LCO 3.8.3.E and LCO 3.8.1.B.

Event 3 - Division 2 ECCS initiation on spurious RPV low level signal (Triggered by Lead Examiner)

When Tech Specs have been addressed, a spurious Division 2 ECCS initiation on low RPV level will occur. The BOP will verify the initiation is spurious by two independent means and recover from the Division 2 ECCS initiation using 04-1-01-E12-1, Residual Heat Removal System SOI, Attachment IX. CRS will enter 05-1-02-I-4, Loss of AC Power. The ATC will recognize the Division 2 half-scram due to RPS Bus B loss of power. CRS will enter 05-1-02-III-2, Loss of One or Both RPS Buses ONEP. BOP will restore RPS Bus B to normal power supply and the ATC will reset the Division 2 half-scram. The CRS will enter LCO 3.3.5.1.B, 3.3.5.1.F, 3.3.6.1.A, B, and F, 3.3.6.3.B and 3.3.6.4.B.

Event 4 - Loss of condenser vacuum (Triggered by Lead Examiner)

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

Event 5/6 - LOP/LOCA/HPCS Pump trip (Automatically triggered)

When the reactor is scrammed, a total loss of offsite power occurs, followed by a small recirculation pipe break after 5 minutes. HPCS pump will trip when it is initiated **(Event 6)**. The CRS enters EP-2 and EP-3. RPV level will lower due to the leak being greater than the capacity of RCIC. When the CRS determines there are insufficient high pressure injection sources to maintain RPV level above -160” wide range, enters Alternate Level Control contingency of EP-2. ATC will inhibit ADS to prevent automatic operation **(CT-1)**. When RPV level lowers to -160” wide range, the crew will emergency depressurize the RPV using ADS/SRVs **(CT-2)** and restore RPV level with Division 1 ECCS systems.

Event 7 - Failure of Division 1 ECCS to automatically initiate (Automatically triggered)

Division 1 ECCS will fail to automatically initiate on either high drywell pressure or low RPV level. ATC will manually initiate Division 1 ECCS using the lock-collared pushbutton **(CT-3)** IAW EN-OP-200, Plant Transient Response Rules.

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

Critical Task	**(CT-1) Inhibit ADS prior to automatic ADS valve opening during a LOCA**	**(CT-2) When RPV level lowers to -160" wide range and cannot be maintained above -191" CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191" CFZ (Momentary shrink below -191" due to automatic SRV closure does not constitute failure of this critical task)**
EVENT	6	6
Safety Significance	Permitting automatic ADS initiation may be undesirable for the following reasons: • ADS actuation can impose a severe thermal transient on the RPV and may complicate efforts to control RPV water level. • If only steam-driven systems are available for injection, ADS actuation may directly lead to loss of adequate core cooling and subsequent core damage. • The conditions assumed in the design of the ADS actuation logic (e.g., no operator action for 115 seconds after event initiation) may not exist when the actions specified in this step are being performed. • The operating crew can draw on much more information than is available to the ADS logic (e.g., equipment out of service for maintenance, operating experience with certain systems, probability of restoration of off-site power, etc.) and can better judge, based on instructions contained in the EPGs/SAGs, when and how to depressurize the RPV. Defeating the logic relieves the operating crew of the task of detecting timer initiation during execution of the more complex steps of Contingency #1 and precludes unnecessary and unwanted automatic initiations. Subsequent steps provide explicit and detailed instructions for controlling RPV water level and specify when emergency depressurization is appropriate.	The MSCWL is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. When water level decreases below MSCWL with injection, clad temperatures may exceed 1500°F.
Cueing	Step L-5 of EP-2, RPV CONTROL, Alternate Level Control Contingency	Wide range indication (SPDS and PDS) falls to -160" and lowering trend continues, and, before -160" wide range is reached, initial conditions, field reports, and control room indications convey that adequate high pressure injection cannot be restored before level falls below -191" CFZ.
Performance Indicator	Manipulation of ADS A and ADS B MANUAL INHIBIT switches on panel P601 vertical section.	Manipulation of seven of the eight ADS/SRVs on panel P601: B21-F041K B21-F047L B21-F041F B21-F047A B21-F051C B21-F041D B21-F051A B21-F051B
Performance Feedback	Inhibit switches click into INHIBIT position on panel P601 vertical section. White indicating light on ADS A and ADS B MANUAL INHIBIT switches illuminate. Receipt of ADS/SRV A and ADS/SRV B OOSVC alarms on panel P601/18A-H2 and P601/19A-H2.	Crew will observe ADS/SRV light indication go from green to red, reactor pressure lowering on SPDS and panel P601 indications.

Justification for the chosen performance limit	The 115 second ADS timer allows sufficient time for the crew to recognize and override automatic operation of the system. As long as ADS is inhibited before ADS valves open, reactor pressure will not be reduced.	The MSCWL (-191” CFZ) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Emergency depressurization is allowed when level goes below TAF (-160” wide range) and should be performed, if in the judgment of the CRS, level cannot be maintained above -191” CFZ. Since it is intended for the scenario supporting this CT to, early in the event, clearly indicate no high pressure injection systems can be made available to reverse the lowering level trend, the crew will have time to communicate and opens at least seven ADS/SRVs before -191” CFZ.
BWR Owners Group Appendix	App. B, step C1-1	App. B, Contingency #1 Step C1-4
License Bases Documents	02-S-01-40, EP Technical Bases, Attachment IV, Step L-5 UFSAR Chapter 15A.6.3.1	02-S-01-40, EP Technical Bases, Attachment IV, Step L-7 – through L-13 UFSAR Chapter 15A.6.3.1

*** If an operator or the crew significantly deviates from, or fails to, follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review (NUREG 1021, Appendix D)**

Critical Task	**(CT-3) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually initiates or aligns ECCS systems for injection prior to RPV pressure lowering below 300 psig**	
EVENT	7	
Safety Significance	Failure to recognize the auto initiation not occurring, and failure to take manual action per Conduct of Ops will result in unavailability of safety-related equipment necessary to provide adequate core cooling, otherwise resulting in core damage and a large offsite release.	
Cueing	Indication of ECCS systems not initiating with initiation conditions present: • Indication of Drywell pressure ≥ 1.39 psig or RPV level ≤ -150.3” wide range • White light on LPCS/RHR A INIT RESET pushbutton extinguished on panel P601 • Green light on and red light extinguished on LPCS and RHR A pump handswitches on panel P601	
Performance Indicator	Operator manually initiates Division 1 ECCS by rotating the arming collar and depressing the LPCS/RHR A MAN INIT pushbutton on panel P601.	
Performance Feedback	Red light on and green light extinguished on LPCS and RHR A pump handswitches on panel P601. Rising level trend on indications on panel P601, PDS and SPDS. Rising flow rate on LPCS and/or RHR A flow indicators on panel P601, PDS, and SPDS.	
Justification for the chosen performance limit	Attempting to align high pressure ECCS systems must be performed to determine their availability by the time TAF is reached in order to properly implement EP-2 decision steps regarding restoring and maintaining RPV level. Attempting to align low pressure ECCS systems can only be done one RPV pressure falls below the injection valve RPV pressure permissive and will only be effective once RPV pressure falls below the shutoff head of the respective ECCS pump. The reduction in RPV pressure will normally be via Emergency Depressurization, which is a separate critical task bounded by a minimum RPV level.	
BWR Owners Group Appendix	App. B, Contingency 1, step C1-3	
License Bases Documents	02-S-01-40, EP Technical Bases, Attachment IV, Step L-14 UFSAR Chapter 15A.6.3.1	

*** If an operator or the crew significantly deviates from, or fails to, follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review (NUREG 1021, Appendix D)**

Simulator Setup:

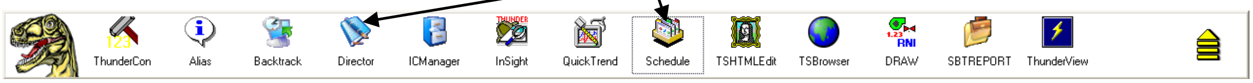
A. Initialization

1. Log off all simulator PDS and SPDS computers (PDS and SPDS must come up after the simulator load for proper operation).

2. Startup the simulator using Simulator Instructor's Job Aid section 7.3.

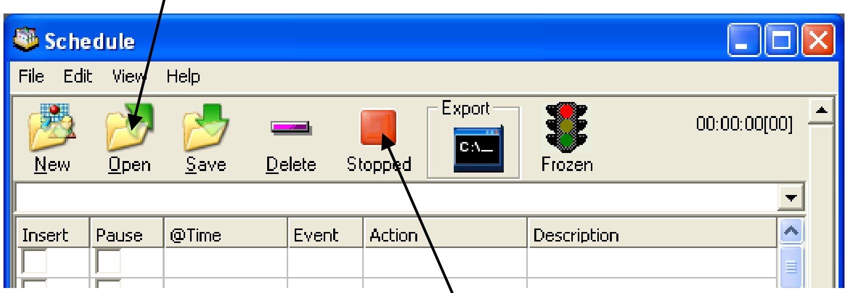
Note:
Prior to running the Schedule File, ensure no Event Files are Open. *If an existing Event File is Open prior to running the Schedule File, then any associated Event Files will not automatically load.*

3. Open Schedule.exe and Director.exe by clicking on the Icon in the Thunder Bar.



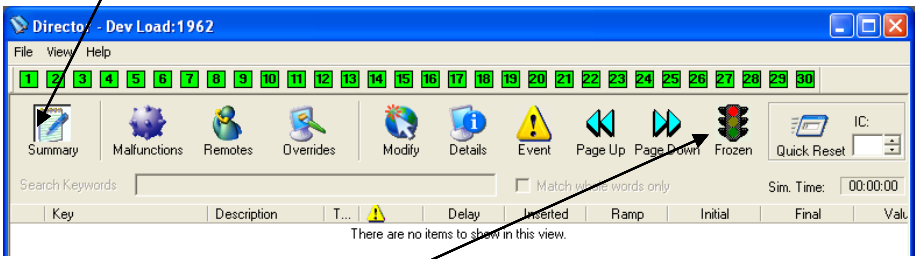
4. Set the Simulator to **IC-102** and perform switch check (Using Quick Reset in Director).

5. Click on "**Open**" in the Schedule window and Open Schedule File "**12-2017 NRC Exam Scenario 3.sch**" (in the Schedule Directory)



6. In Schedule window, click on the "**Stopped**" red block. The red block will change to a green arrow and indicate the scenario is active ("**Running**").

7. Click the Summary tab in the Director window. Verify the schedule files are loaded and opened per Section B below. (Note: Any actions in the schedule file without a specific time will not load into the director until triggered.)



8. Take the simulator out of freeze.

9. Log on to all simulator PDS and SPDS computers.

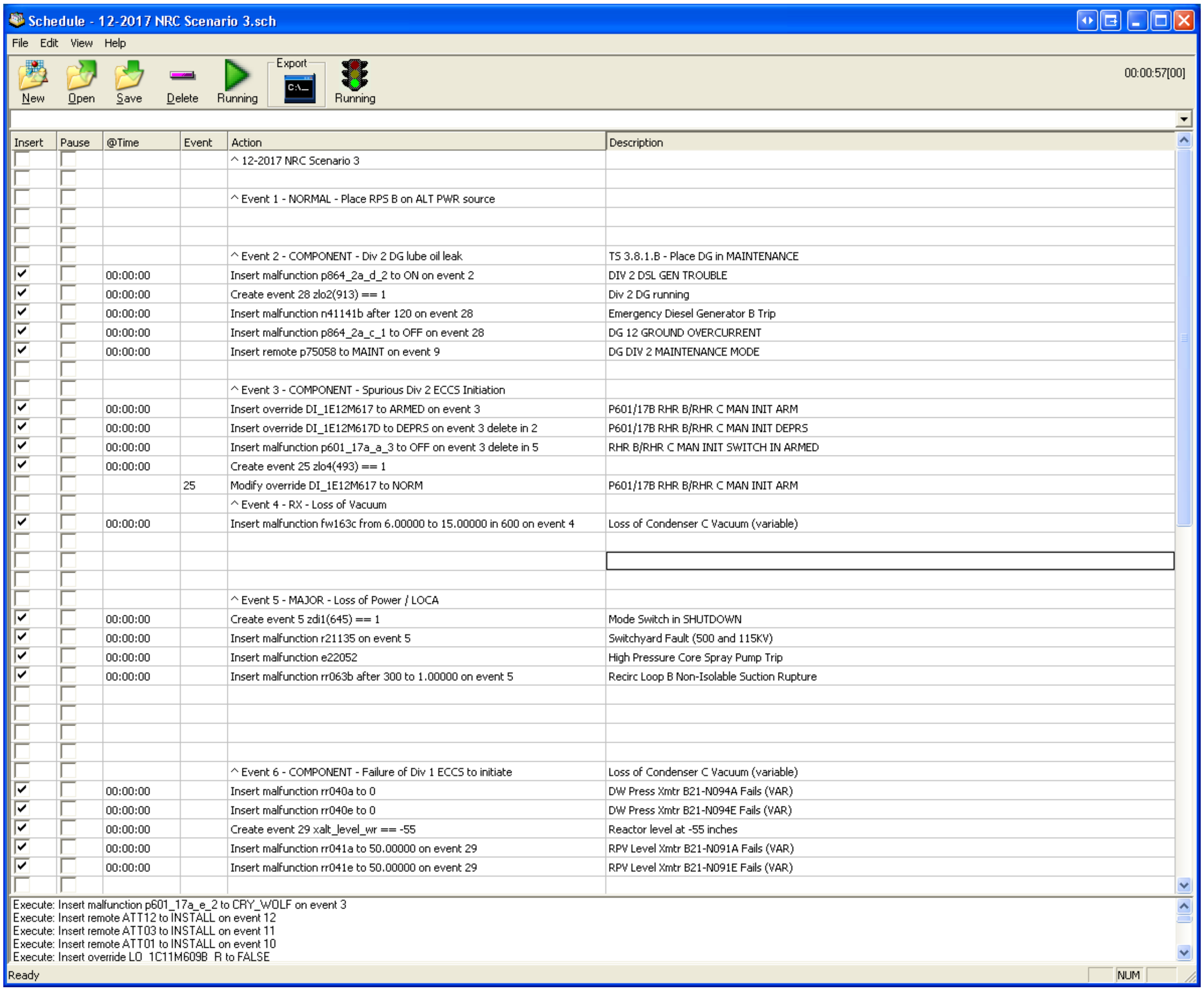
10. Verify or perform the following:

- IC-102
- Place red tag on TBCW C pump handswitch
- Place red tag on CRD PMP B and CRD PUMP B AUX OIL PUMP handswitches
- Ensure all procedures are marked as indicated for turnover conditions
- Advance all chart recorders and ensure all pens are inking properly
- Clear any graphs and trends off PDS and SPDC

11. Run through any alarms and ensure alarms are on. **(Note: On T-Rex, to verify alarms are ON, the indicator will indicate “Alarms On”).**

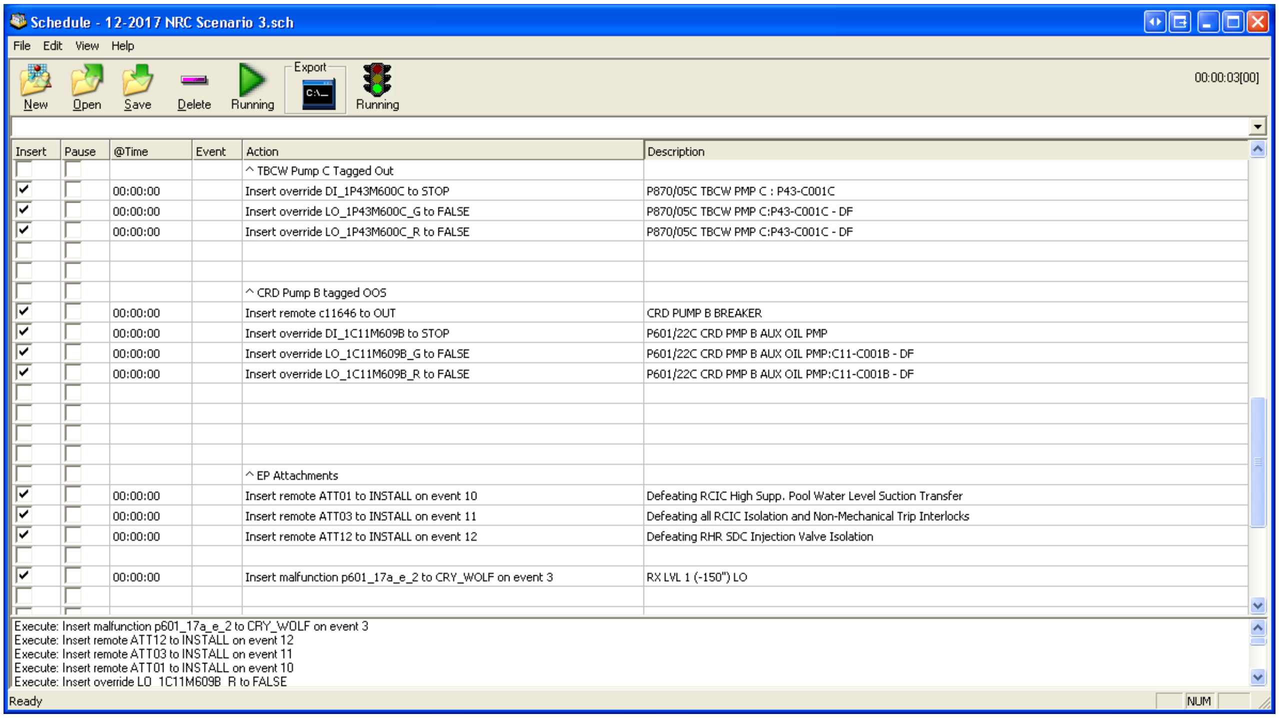
12. Place the simulator in Freeze.

File loaded verification:



Insert	Pause	@Time	Event	Action	Description
				^ 12-2017 NRC Scenario 3	
				^ Event 1 - NORMAL - Place RPS B on ALT PWR source	
				^ Event 2 - COMPONENT - Div 2 DG lube oil leak	TS 3.8.1.B - Place DG in MAINTENANCE
✔		00:00:00		Insert malfunction p864_2a_d_2 to ON on event 2	DIV 2 DSL GEN TROUBLE
✔		00:00:00		Create event 28 zlo2(913) == 1	Div 2 DG running
✔		00:00:00		Insert malfunction n41141b after 120 on event 28	Emergency Diesel Generator B Trip
✔		00:00:00		Insert malfunction p864_2a_c_1 to OFF on event 28	DG 12 GROUND OVERCURRENT
✔		00:00:00		Insert remote p75058 to MAINT on event 9	DG DIV 2 MAINTENANCE MODE
				^ Event 3 - COMPONENT - Spurious Div 2 ECCS Initiation	
✔		00:00:00		Insert override DI_1E12M617 to ARMED on event 3	P601/17B RHR B/RHR C MAN INIT ARM
✔		00:00:00		Insert override DI_1E12M617D to DEPRS on event 3 delete in 2	P601/17B RHR B/RHR C MAN INIT DEPRS
✔		00:00:00		Insert malfunction p601_17a_a_3 to OFF on event 3 delete in 5	RHR B/RHR C MAN INIT SWITCH IN ARMED
✔		00:00:00		Create event 25 zlo4(493) == 1	
			25	Modify override DI_1E12M617 to NORM	P601/17B RHR B/RHR C MAN INIT ARM
				^ Event 4 - RX - Loss of Vacuum	
✔		00:00:00		Insert malfunction fw163c from 6.00000 to 15.00000 in 600 on event 4	Loss of Condenser C Vacuum (variable)
				^ Event 5 - MAJOR - Loss of Power / LOCA	
✔		00:00:00		Create event 5 zdi1(645) == 1	Mode Switch in SHUTDOWN
✔		00:00:00		Insert malfunction r21135 on event 5	Switchyard Fault (500 and 115KV)
✔		00:00:00		Insert malfunction e22052	High Pressure Core Spray Pump Trip
✔		00:00:00		Insert malfunction rr063b after 300 to 1.00000 on event 5	Recirc Loop B Non-Isolable Suction Rupture
				^ Event 6 - COMPONENT - Failure of Div 1 ECCS to initiate	Loss of Condenser C Vacuum (variable)
✔		00:00:00		Insert malfunction rr040a to 0	DW Press Xmtr B21-N094A Fails (VAR)
✔		00:00:00		Insert malfunction rr040e to 0	DW Press Xmtr B21-N094E Fails (VAR)
✔		00:00:00		Create event 29 xalt_level_wr == -55	Reactor level at -55 inches
✔		00:00:00		Insert malfunction rr041a to 50.00000 on event 29	RPV Level Xmtr B21-N091A Fails (VAR)
✔		00:00:00		Insert malfunction rr041e to 50.00000 on event 29	RPV Level Xmtr B21-N091E Fails (VAR)

Execute: Insert malfunction p601_17a_e_2 to CRY_WOLF on event 3
Execute: Insert remote ATT12 to INSTALL on event 12
Execute: Insert remote ATT03 to INSTALL on event 11
Execute: Insert remote ATT01 to INSTALL on event 10
Execute: Insert override LO 1C11M609B R to FALSE
Ready NUM



Schedule - 12-2017 NRC Scenario 3.sch

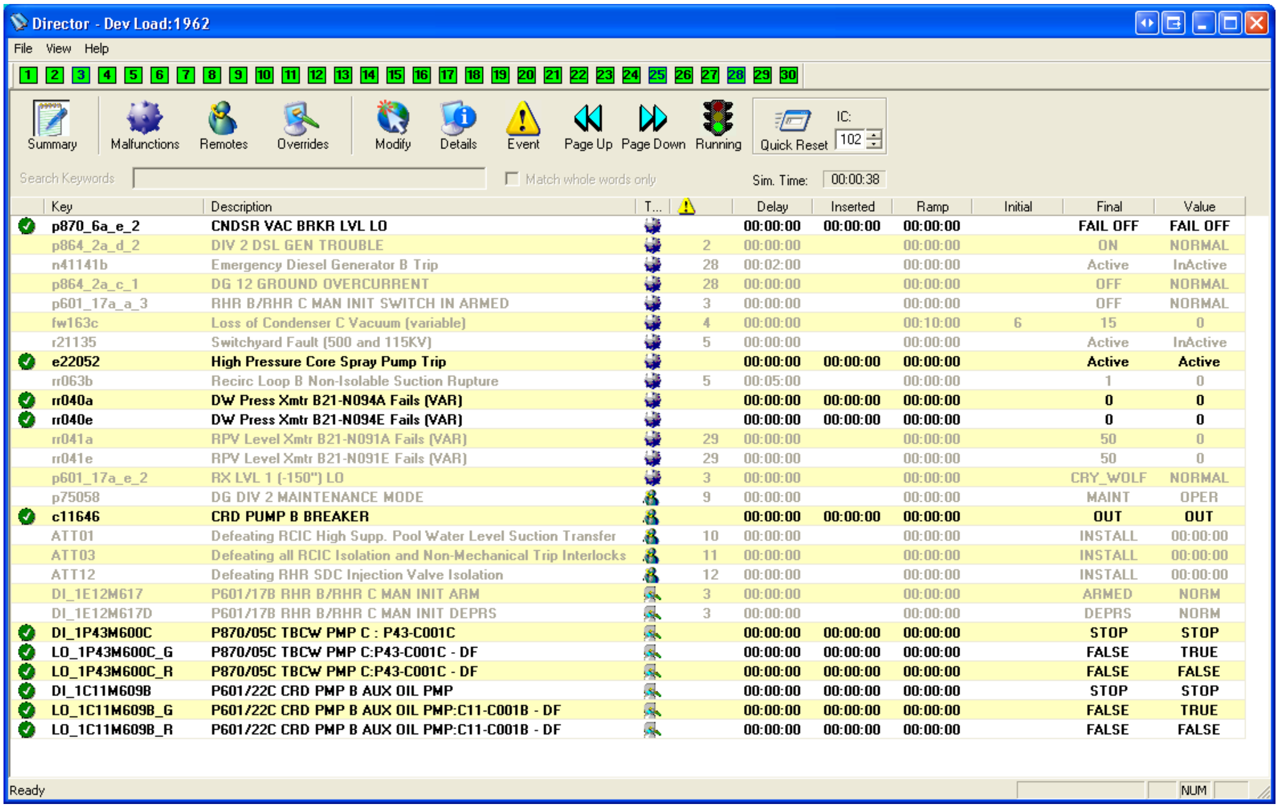
File Edit View Help

New Open Save Delete Running Export Running 00:00:03[00]

Insert	Pause	@Time	Event	Action	Description
				^ TBCW Pump C Tagged Out	
✓		00:00:00		Insert override DI_1P43M600C to STOP	P870/05C TBCW PMP C : P43-C001C
✓		00:00:00		Insert override LO_1P43M600C_G to FALSE	P870/05C TBCW PMP C:P43-C001C - DF
✓		00:00:00		Insert override LO_1P43M600C_R to FALSE	P870/05C TBCW PMP C:P43-C001C - DF
				^ CRD Pump B tagged OOS	
✓		00:00:00		Insert remote c11646 to OUT	CRD PUMP B BREAKER
✓		00:00:00		Insert override DI_1C11M609B to STOP	P601/22C CRD PMP B AUX OIL PMP
✓		00:00:00		Insert override LO_1C11M609B_G to FALSE	P601/22C CRD PMP B AUX OIL PMP:C11-C001B - DF
✓		00:00:00		Insert override LO_1C11M609B_R to FALSE	P601/22C CRD PMP B AUX OIL PMP:C11-C001B - DF
				^ EP Attachments	
✓		00:00:00		Insert remote ATT01 to INSTALL on event 10	Defeating RCIC High Supp. Pool Water Level Suction Transfer
✓		00:00:00		Insert remote ATT03 to INSTALL on event 11	Defeating all RCIC Isolation and Non-Mechanical Trip Interlocks
✓		00:00:00		Insert remote ATT12 to INSTALL on event 12	Defeating RHR SDC Injection Valve Isolation
✓		00:00:00		Insert malfunction p601_17a_e_2 to CRY_WOLF on event 3	RX LVL 1 (-150") LO

Execute: Insert malfunction p601_17a_e_2 to CRY_WOLF on event 3
Execute: Insert remote ATT12 to INSTALL on event 12
Execute: Insert remote ATT03 to INSTALL on event 11
Execute: Insert remote ATT01 to INSTALL on event 10
Execute: Insert override LO 1C11M609B R to FALSE

Ready NUM



Director - Dev Load:1962

File View Help

1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30

Summary Malfunctions Remotes Overrides Modify Details Event Page Up Page Down Running Quick Reset IC: 102

Search Keywords Match whole words only Sim. Time: 00:00:38

Key	Description	T...	⚠	Delay	Inserted	Ramp	Initial	Final	Value
p870_6a_e_2	CNDSR VAC BRKR LVL LO			00:00:00	00:00:00	00:00:00		FAIL OFF	FAIL OFF
p864_2a_d_2	DIV 2 DSL GEN TROUBLE		2	00:00:00		00:00:00		ON	NORMAL
n41141b	Emergency Diesel Generator B Trip		28	00:02:00		00:00:00		Active	InActive
p864_2a_c_1	DG 12 GROUND OVERCURRENT		28	00:00:00		00:00:00		OFF	NORMAL
p601_17a_a_3	RHR B/RHR C MAN INIT SWITCH IN ARMED		3	00:00:00		00:00:00		OFF	NORMAL
fw163c	Loss of Condenser C Vacuum (variable)		4	00:00:00		00:10:00	6	15	0
r21135	Switchyard Fault (500 and 115KV)		5	00:00:00		00:00:00		Active	InActive
e22052	High Pressure Core Spray Pump Trip			00:00:00	00:00:00	00:00:00		Active	Active
rr063b	Recirc Loop B Non-Isolable Suction Rupture		5	00:05:00		00:00:00		1	0
rr040a	DW Press Xmtr B21-N094A Fails (VAR)			00:00:00	00:00:00	00:00:00		0	0
rr040e	DW Press Xmtr B21-N094E Fails (VAR)			00:00:00	00:00:00	00:00:00		0	0
rr041a	RPV Level Xmtr B21-N091A Fails (VAR)		29	00:00:00		00:00:00		50	0
rr041e	RPV Level Xmtr B21-N091E Fails (VAR)		29	00:00:00		00:00:00		50	0
p601_17a_e_2	RX LVL 1 (-150") LO		3	00:00:00		00:00:00		CRY_WOLF	NORMAL
p75058	DG DIV 2 MAINTENANCE MODE		9	00:00:00		00:00:00		MAINT	OPER
c11646	CRD PUMP B BREAKER			00:00:00	00:00:00	00:00:00		OUT	OUT
ATT01	Defeating RCIC High Supp. Pool Water Level Suction Transfer		10	00:00:00		00:00:00		INSTALL	00:00:00
ATT03	Defeating all RCIC Isolation and Non-Mechanical Trip Interlocks		11	00:00:00		00:00:00		INSTALL	00:00:00
ATT12	Defeating RHR SDC Injection Valve Isolation		12	00:00:00		00:00:00		INSTALL	00:00:00
DI_1E12M617	P601/17B RHR B/RHR C MAN INIT ARM		3	00:00:00		00:00:00		ARMED	NORM
DI_1E12M617D	P601/17B RHR B/RHR C MAN INIT DEPRS		3	00:00:00		00:00:00		DEPRS	NORM
DI_1P43M600C	P870/05C TBCW PMP C : P43-C001C			00:00:00	00:00:00	00:00:00		STOP	STOP
LO_1P43M600C_G	P870/05C TBCW PMP C:P43-C001C - DF			00:00:00	00:00:00	00:00:00		FALSE	TRUE
LO_1P43M600C_R	P870/05C TBCW PMP C:P43-C001C - DF			00:00:00	00:00:00	00:00:00		FALSE	FALSE
DI_1C11M609B	P601/22C CRD PMP B AUX OIL PMP			00:00:00	00:00:00	00:00:00		STOP	STOP
LO_1C11M609B_G	P601/22C CRD PMP B AUX OIL PMP:C11-C001B - DF			00:00:00	00:00:00	00:00:00		FALSE	TRUE
LO_1C11M609B_R	P601/22C CRD PMP B AUX OIL PMP:C11-C001B - DF			00:00:00	00:00:00	00:00:00		FALSE	FALSE

Ready NUM

Procedures that may be used in this scenario:

Procedure No.	Rev	Procedure Title
04-1-01-C11-1	154	Control Rod Drive Hydraulic System
04-1-01-C41-1	123	Standby Liquid Control System
04-1-01-C71-1	35	Reactor Protection System
04-1-01-E12-1	147	Residual Heat Removal System
04-1-01-E30-1	25	Suppression Pool Makeup System
04-1-01-E51-1	136	Reactor Core Isolation Cooling System
04-1-01-E61-1	41	Combustible Gas Control System
04-1-01-P75-1	106	Standby Diesel Generator System
04-1-02-1H13-P601	161	Alarm Response Instruction Panel No.: 1H13-P601
04-1-02-1H13-P680	233	Alarm Response Instruction Panel No.:1H13-P680
04-1-02-1H13-P864	31	Alarm Response Instruction Panel No.: 1H13-P864
04-1-02-1H13-P870	154	Alarm Response Instruction Panel No.: 1H13-P870
04-S-02-SH13-P807	32	Alarm Response Instruction Panel No.: SH13-P807
04-1-02-1H22-P401	118	Alarm Response Instruction Panel No.: 1H22-P401
05-1-02-I-1	130	Reactor Scram
05-1-02-I-2	37	Turbine and Generator Trips
05-1-02-I-4	51	Loss of AC Power
05-1-02-III-2	26	Loss of One or Both RPS Buses
05-1-02-III-3	115	Reduction In Recirculation System Flow Rate
05-1-02-III-5	49	Automatic Isolations
05-1-02-V-1	24	Loss of Component Cooling Water
05-1-02-V-8	24	Loss of Condenser Vacuum
05-S-01-EP-1	36	Emergency / Severe Accident Procedure Support Documents
05-S-01-EP-2	45	RPV Control
05-S-01-EP-3	29	Containment Control
Tech Spec 3.3.5.1		
Tech Spec 3.3.6.1		
Tech Spec 3.3.6.3		
Tech Spec 3.3.6.4		
Tech Spec 3.8.1		
Tech Spec 3.8.3		

Expected Alarms:

P680-7A-A2, RX SCRAM TRIP
P864-2A-D2, DIV 2 DSL GEN TROUBLE
P864-2A-B1, DIV 2 DSL GEN TRIP
P864-2A-D1, DG 12 AUTO START NOT AVAIL
P601-17A-D2 RHR PMP B AUTO START
P601-17A-H3 RHR C SYS OOSVC
P601-17A-B3 RHR B-RHR C ACTUATED
P807-3A-H4 STATIC INVRTR 1Y97 TROUBLE
P807-3A-H3 STATIC INVRTR 1Y82 TROUBLE
P807-3A-H2 STATIC INVRTR 1Y81 TROUBLE
P601-17A-D5 RHR PMP C AUTO START
P870-8A-E1 CCW PMP B DISCH PRESS LO
P870-8A-A1 CCW PMP B TRIP
P870-5A-C2 CCW PMP A-C DISCH PRESS LO
P680-4A2-B6 FPCC FLTR DMIN SYS TROUBLE
P845-1A-A4 ADSORBER TRAIN A FLOW HIGH-LOW
P845-1A-B4 ADSORBER TRAIN B FLOW HIGH-LOW
P680-4A2-E3 OG PNL P845 TROUBLE
P680-10A-C9 LP CNDSR SHELL PRESS HI
P680-10A-A8 TURB VAC LO

INITIAL CONDITIONS

A. Plant Status: 100% power, middle of cycle

B. Tech. Spec. Limitations in effect: None

C. Significant problems/abnormalities:

1. TBCW Pump C is tagged out for motor oil replacement.
2. CRD Pump B is tagged out of service for oil replacement in the speed increaser.

D. Integrated Risk: Green

E. Division Work Week: Division 2

F. Evolutions/maintenance for the up-coming shift :

1. Transfer RPS Bus B from normal to alternate power supply in preparation for maintenance on the RPS B Motor Generator.
2. The Motor Generator will be tagged out on the next shift.
3. No scram or isolation surveillances are in progress or planned for this shift.

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

Examiners: ______________ Operators: ______________

Event No.	Malf. No.	Event Type †	Event Description
1	N/A	N (ATC,CRS)	Withdraw control rods to 10% Bypass Valve position
2	fw115c	C (ATC,CRS) A (CREW)	Condensate Pump C trip
3	pte22n654c_a	TS (CRS) I (BOP,CRS)	HPCS CST Level Lo trip unit failing high
4	fw274	C(ATC,CRS) A(CREW)	Startup Level Control controller failing downscale (Feedwater Malfunctions ONEP)
5	r21142u r21134g n41140a	TS (CRS) C(BOP,CRS) A (CREW)	ESF Transformer 11 trip with failure of Div 1 DG to start and failure 15BA4 to re-energize (Loss of AC Power ONEP)
6	ms066a ms183a ms184a	M (CREW)	Main Steam Tunnel steam leak with failure of one steam line to isolate * **(CT-1) When MSIVs fail to isolate, manually scram the reactor and close the MSIVs prior to Steam Tunnel temperature exceeding 250°F (Max Safe Temperature)**
7	b21f065b_i fw171b	M (CREW)	Feedwater Line B line break inside Drywell with B21-F065B power loss
8	e22052	C (ATC,BOP,CRS)	HPCS Pump Trip * **(CT-2) Inhibit ADS prior to automatic ADS valve opening during a LOCA** * **(CT-3) When RPV level lowers to -160” wide range and cannot be maintained above -191” CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191” CFZ (Momentary shrink below -191” due to automatic SRV closure does not constitute failure of this critical task)**
9	r21219	C (BOP,CRS)	LPCS logic power failure * **(CT-4) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually aligns ECCS systems for injection prior to RPV pressure lowering below 300 psig**

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

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

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

Reason for Revision: **Editorial changes to enhance Technical Specification actions.**

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

1. Withdraw control rods to 10% Bypass Valve position
2. Respond to Condensate Pump B trip.
3. Respond to a HPCS CST Level LO trip unit failing high.
4. Respond to a Startup Level Control controller failing downscale.
5. Respond to an ESF Transformer 11 trip with a failure of Div 1 DG and failure of 15BA4 to re-energize.
6. Respond to an Main Steam Tunnel steam leak with failure of one steam line to isolate.
7. Respond to a Feedwater Line B line break inside the Drywell with B21-F065B power loss.
8. Respond to a HPCS Pump trip.
9. Respond to a LPCS logic power failure.

Initial Conditions:

- Reactor power is approximately 4% power
- Reactor pressure is 750 psig

Inoperable Equipment: None

Turnover:

- Reactor startup is in progress:
  - Step 45 of IOI 03-1-01-1, Attachment XV
  - Step 96 of Control Rod Movement Sequence is complete
  - SJAE B is in service
- Condensate System is lined up as follows:
  - Condensate Pumps A and C in service
  - Condensate Booster Pump C in service
  - Reactor Feed Pump A in service at approximately 950 psig discharge pressure
  - CFFF is in service
  - 4 Deep-Bed Condensate Demineralizers are in service
- Annunciators P680-4A2-C5, CONT ROD WITHDRAWAL BLOCK, and P680-4A1-A7, CRD DRIVE WTR TO RX ΔP HI, are flagged as expected annunciators

Planned activities for this shift are:

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

Scenario Notes:

This scenario is a NEW Scenario.

Validation Time: 60 minutes

Quantitative Attributes Table			
Attribute	**E3-304-1 Target**	**Actual**	**Description**
Malfunctions after EOP entry	1-2	2	• HPCS Pump Trip E8) • LPCS Logic power failure (E9)
Abnormal Events	2-4	3	• Condensate Pump C trip (Feedwater Malfunctions ONEP) (E2) • Startup Level Control Controller fails downscale (Feedwater Malfunctions ONEP) (E4) • ESF Transformer 11 trip with failure of Div 1 DG to start and failure 15BA4 to re-energize (Loss of AC Power ONEP) (E5)
Major Transients	1-2	2	• Main Steam Tunnel steam leak with failure of one steam line to isolate (E6) • Feedwater Line B line break inside the Drywell with B21-F065B power loss (E7)
EOP entries requiring substantive action	1-2	3	• EP-4 • EP-2 • EP-3
EOP contingencies requiring substantive action	0-2	2	• EP-2 Alternate Level Control • EP-2 Emergency Depressurization
EOP based Critical Tasks	2-3	4	• **(CT-1) When MSIVs fail to isolate, manually scram the reactor and close the MSIVs prior to Steam Tunnel temperature exceeding 250°F (Max Safe Temperature)** • **(CT-2) Inhibit ADS prior to automatic ADS valve opening during a LOCA** • **(CT-3) When RPV level lowers to -160" wide range and cannot be maintained above -191" CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191" CFZ. (Momentary shrink below -191" due to automatic SRV closure does not constitute failure of this critical task.)** • **(CT-4) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually aligns ECCS systems for injection prior to RPV pressure lowering below 300 psig**
Normal Events	N/A	1	• Withdraw control rods to 10% Bypass Valve position (E1)
Reactivity Manipulations	N/A	0	• N/A
Instrument / Component failures	N/A	6	• Condensate Pump C trip (E2) • HPCS CST Level LO trip unit failing high (E3) • Startup Level Control controller failing downscale (E4) • ESF Transformer 11 trip with failure of Div 1 DG to start and failure 15BA4 to re-energize (Loss of AC Power ONEP) (E5) • HPCS Pump trip (E8) • LPCS logic power failure (E9)

Total Malfunctions	N/A	8	• Condensate Pump C trip (E2) • HPCS CST Level LO trip unit failing high (E3) • Startup Level Control controller failing downscale (E4) • ESF Transformer 11 trip with failure of Div 1 DG to start and failure 15BA4 to re-energize (Loss of AC Power ONEP) (E5) • Main Steam Tunnel steam leak with failure of one steam line to isolate (E6) • Feedwater Line B line break inside the Drywell with a B21-F065B power loss (E7) • HPCS pump trip (E8) • LPCS logic power failure (E9)

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

ADS (Event 7)
RHR (Event 9)
ESF Power (R20) (Event 5)
Condensate (Event 2)
Failure to manually depressurize with ADS/SRVs (Event 8)
Failure to align alternate power to 4.16 KV or 6.9KV buses (Event 5)

SCENARIO ACTIVITIES:

The plant is operating at 4% power during a reactor startup.

Event 1 - Withdraw control rods to 10% Bypass Valve position

After the crew assumes the shift, the ATC will withdraw control rods in accordance with Control Rod Movement Sequence IAW SOI 04-1-01-C11-2, Rod Control and Information System.

Event 2 – Condensate Pump C trip (Triggered by Lead Examiner)

At the direction of the Lead Examiner, Condensate Pump C will trip. CRS will enter 05-1-02-V-7, Feedwater System Malfunctions ONEP, and direct ATC to start Condensate Pump B IAW SOI 04-1-01-N19-1.

Event 3 - HPCS CST Level Lo trip unit failing upscale (Triggered by Lead Examiner)

After actions of Condensate Pump C trip are complete, HPCS CST Level LO trip unit, E22-N654C, will fail high. CRS will enter LCO 3.3.5.1 Condition A and, using Table 3.3.5.1-1, enter LCO 3.3.5.1 Condition D. CRS will direct BOP to transfer HPCS suction from CST to Suppression Pool IAW SOI 04-1-01-E22-1.

Event 4 - Startup Level Control controller failing downscale (Triggered by Lead Examiner)

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

Event 5 – ESF Transformer 11 trip with failure of 15BA4 to re-energize (Triggered by Lead Examiner)

After RPV level is stabilized, ESF Transformer 11 will trip. Division 1 Diesel Generator will fail to start. CRS will enter 05-1-02-I-4, Loss of AC Power ONEP, 05-1-02-III-5, Automatic Isolations ONEP and 05-1-02-III-1, Inadequate Decay Heat Removal ONEP. BOP will re-energize Bus 15AA from ESF Transformer 12 and restore Instrument Air to CTMT by opening P53-F001. BOP will recognize the failure of 15BA4 to re-energize. CRS will enter LCO 3.8.7, Condition A, for LCC 15BA4 failure and LCO 3.8.1.B for failure of Division 1 Diesel Generator.

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

Event 6 - Main Steam Tunnel steam leak with failure of one steam line to isolate (Triggered by Lead Examiner)

When ESF Bus 15AA has been re-energized and Tech Specs addressed, a steam leak in the Auxiliary Building Main Steam Tunnel will occur. The 'A' Steam Line will fail to isolate. The CRS will enter EP-4 and direct the ATC to manually scram the reactor and the BOP to manually close B21-F022A, INBD MSIV, and B21-F028A, OTBD MSIV **(CT-1)**. When the reactor is scrammed, the CRS will enter EP-2.

Event 7 - Feedwater Line B line break inside Drywell with B21-F065B power loss (Triggered automatically)

When the reactor is scrammed, an unisolable Feedwater Line 'B' break in the Drywell will occur. The BOP will secure all Condensate Pumps and close B21-F065B, FW INL SHUTOFF VLV. B21-F065B will not close due to a power loss when its CLOSE handswitch is depressed.

Event 8 – HPCS Pump Trip (Triggered automatically)

When Drywell pressure reaches 1.39 psig, HPCS Pump will trip and ESF Bus 16AB will lockout after 5 minutes causing a loss of all Division 2 ECCS. When CRS determines there are insufficient high pressure injection sources to maintain RPV level above -160" wide, enters Alternate Level Control contingency of EP-2. Crew will inhibit ADS to prevent automatic operation **(CT-2)**. When RPV level lowers to -160" wide range, the crew will emergency depressurize the RPV using ADS/SRVs **(CT-3)** and restore RPV level with Division 1 ECCS systems.

Event 9 - LPCS logic power failure (Triggered automatically)

When Drywell pressure rises to 1.39 psig, a LPCS logic power failure will occur. BOP will respond using ARI 04-1-02-1H13-P601-21A-H8, LPCS SYS OOSVC, and manually align Div 1 ECCS systems for injection to the RPV **(CT-4)**.

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

Critical Task	**(CT-1) When MSIVs fail to isolate, manually scram the reactor and close the MSIVs prior to Steam Tunnel temperature exceeding 250°F (Max Safe Temperature)**	**(CT-2) Inhibit ADS prior to automatic ADS valve opening during a LOCA**
EVENT	6	8
Safety Significance	If a primary system is discharging into the secondary containment when this step of the procedure is reached, one of three conditions must exist: • A primary system break cannot be isolated because system operation is required to assure adequate core cooling or to shut down the reactor. • No isolation valves exist upstream of a primary system break, or if isolation valves do exist, they cannot be closed because of some mechanical/ electrical/pneumatic failure. • The source of the discharge cannot be determined. Since the RPV is the only significant source of heat, other than a fire, which might cause area temperatures to increase to their maximum safe operating values, the action of manually scramming the reactor should terminate increasing secondary containment temperatures. If temperatures in any one of the areas listed in Table SC-1 of the Secondary Containment Control guideline approach their maximum safe operating value, adequate core cooling, containment integrity, safety of personnel, or continued operability of equipment required to perform EPG actions can no longer be assured.	Permitting automatic ADS initiation may be undesirable for the following reasons: • ADS actuation can impose a severe thermal transient on the RPV and may complicate efforts to control RPV water level. • If only steam-driven systems are available for injection, ADS actuation may directly lead to loss of adequate core cooling and subsequent core damage. • The conditions assumed in the design of the ADS actuation logic (e.g., no operator action for 115 seconds after event initiation) may not exist when the actions specified in this step are being performed. • The operating crew can draw on much more information than is available to the ADS logic (e.g., equipment out of service for maintenance, operating experience with certain systems, probability of restoration of off-site power, etc.) and can better judge, based on instructions contained in the EPGs/SAGs, when and how to depressurize the RPV. Defeating the logic relieves the operating crew of the task of detecting timer initiation during execution of the more complex steps of Contingency #1 and precludes unnecessary and unwanted automatic initiations. Subsequent steps provide explicit and detailed instructions for controlling RPV water level and specify when emergency depressurization is appropriate.
Cueing	Main Steam Tunnel temperature rising on PDS. Main Steam Tunnel temperature alarms on panel P601. MSIV open position indication on panel P601 and panel P858.	Step L-5 of EP-2, RPV CONTROL, Alternate Level Control Contingency
Performance Indicator	Operator places the Reactor Mode Switch to SHUTDOWN on panel P680. Operator manipulates switches for MSIVs for Steam Line ‘A’ to CLOSE on panel P601.	Manipulation of ADS A and ADS B MANUAL INHIBIT switches on panel P601 vertical section.
Performance Feedback	RPS Group lights de-energized on panel P680. Control Rod full –in indication on panel P680. Reactor power trend on nuclear instrumentation on panel P680. Green light indication energized and red light indication off for MSIVs for Steam Line ‘A’ on panel P601 and P858.	Inhibit switches click into INHIBIT position on panel P601 vertical section. White indicating light on ADS A and ADS B MANUAL INHIBIT switches illuminate. Receipt of ADS/SRV A and ADS/SRV B OOSVC alarms on panel P601/18A-H2 and P601/19A-H2.
Justification for the chosen performance limit	If temperatures in any one of the areas listed in Table SC-1 of the Secondary Containment Control guideline approach their maximum safe operating value, adequate core cooling, containment integrity, safety of personnel, or continued operability of equipment required to perform EPG actions can no longer be assured.	The 115 second ADS timer allows sufficient time for the crew to recognize and override automatic operation of the system. As long as ADS is inhibited before ADS valves open, reactor pressure will not be reduced.
BWR Owners Group Appendix	App. B, step SC/T-4 and SC/T-4.1	App. B, step C1-1

Licensed Bases Documents	02-S-01-40, EP Technical Bases, Attachment VII, Step 8 through 10 UFSAR Chapter 15.6.4	02-S-01-40, EP Technical Bases, Attachment IV, Step L-5 UFSAR Chapter 15A.6.5.3

*** If an operator or the crew significantly deviates from, or fails to, follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review (NUREG 1021, Appendix D)**

Critical Task	**(CT-3) When RPV level lowers to -160” wide range and cannot be maintained above -191” CFZ (MSCWL) and insufficient high pressure injection systems are available to restore level, crew begins to Emergency Depressurize by opening at least seven SRVs before RPV level lowers below -191” CFZ (Momentary shrink below -191” due to automatic SRV closure does not constitute failure of this critical task)**	**(CT-4) When operating injection systems cannot maintain RPV level and ECCS systems fail to automatically initiate, crew manually aligns ECCS systems for injection prior to RPV pressure lowering below 300 psig**
EVENT	8	9
Safety Significance	The MSCWL is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. When water level decreases below MSCWL with injection, clad temperatures may exceed 1500°F.	Failure to recognize the auto initiation not occurring, and failure to take manual action per Conduct of Ops will result in unavailability of safety-related equipment necessary to provide adequate core cooling, otherwise resulting in core damage and a large offsite release.
Cueing	Wide range indication (SPDS and PDS) falls to -160” and lowering trend continues, and, before -160” wide range is reached, initial conditions, field reports, and control room indications convey that adequate high pressure injection cannot be restored before level falls below -191” CFZ.	Indication of ECCS systems not initiating with initiation conditions present: • Indication of Drywell pressure ≥ 1.39 psig or RPV level ≤ -150.3” wide range • White light on LPCS/RHR A INIT RESET pushbutton extinguished on panel P601 • Green light on and red light extinguished on LPCS and RHR A pump handswitches on panel P601 • LPCS SYS OOSVC annunciator on panel P601
Performance Indicator	Manipulation of seven of the eight ADS/SRVs on panel P601: B21-F041K B21-F047L B21-F041F B21-F047A B21-F051C B21-F041D B21-F051A B21-F051B	Operator manually manipulates switches for Div 1 ECCS pumps and directs operators to manually open Div 1 ECCS injection valves from Division 1 Remote Shutdown Panel (RHR A) and locally (LPCS).
Performance Feedback	Crew will observe ADS/SRV light indication go from green to red, reactor pressure lowering on SPDS and panel P601 indications.	Red light on and green light extinguished on LPCS and/or RHR A pump and valve handswitches on panel P601. Rising level trend on indications on panel P601, PDS and SPDS. Rising flow rate on LPCS and/or RHR A flow indicators on panel P601, PDS, and SPDS
Justification for the chosen performance limit	The MSCWL (-191” CFZ) is the lowest RPV water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1500°F. Emergency depressurization is allowed when level goes below TAF (-160” wide range) and should be performed, if in the judgment of the CRS, level cannot be maintained above -191” CFZ. Since it is intended for the scenario supporting this CT to, early in the event, clearly indicate no high pressure injection systems can be made available to reverse the lowering level trend, the crew will have time to communicate and open 7 of 8 ADS/SRVs before -191” CFZ.	Attempting to align high pressure ECCS systems must be performed to determine their availability by the time TAF is reached in order to properly implement EP-2 decision steps regarding restoring and maintaining RPV level. Attempting to align low pressure ECCS systems can only be done once RPV pressure falls below the injection valve RPV pressure permissive and will only be effective once RPV pressure falls below the shutoff head of the respective ECCS pump. The reduction in RPV pressure will normally be via Emergency Depressurization, which is a separate critical task bounded by a minimum RPV level.
BWR Owners Group Appendix	App. B, Contingency #1 Step C1-4	App. B, Contingency 1, step C1-3

Licensed Bases Documents	02-S-01-40, EP Technical Bases, Attachment IV, Step L-7 – through L-13 UFSAR Chapter 15A.6.5.3	02-S-01-40, EP Technical Bases, Attachment IV, Step L-14 UFSAR Chapter 15A.6.5.3

*** If an operator or the crew significantly deviates from, or fails to, follow procedures that affect the maintenance of basic safety functions, those actions may form the basis of a CT identified in the post-scenario review (NUREG 1021, Appendix D)**

**** For Crew 1, consisting of I1, R1, and R2, a Post Scenario CT was created because the crew failed to recognize that the ESF 11 trip caused a loss of power to the bus that powers the F001 valve, which is the instrument air to containment isolation valve. The bus loss to this valve was not recognized and the valve was not reopened before control rods started to drift into the core. By procedure when multiple control rods start to drift the crew is required to SCRAM the reactor. The crew recognized that multiple control rods were drifting in and completed the SCRAM of the reactor, however they created this critical task because they unnecessarily SCRAMMED when they were expected to reopen the F0001 valve and restore instrument air, which would have prevented any rod drift conditions and keep the reactor at power. This was an unnecessary challenge on the RPS system when it was not needed. The Chief Examiner consulted with HQ staff to confirm it was a post-scenario CT before the examination team left the site that day and HQ staff concurred that it was a post-scenario CT.**

Simulator Setup:

A. Initialization

1. Log off all simulator PDS and SPDS computers (PDS and SPDS must come up after the simulator load for proper operation).

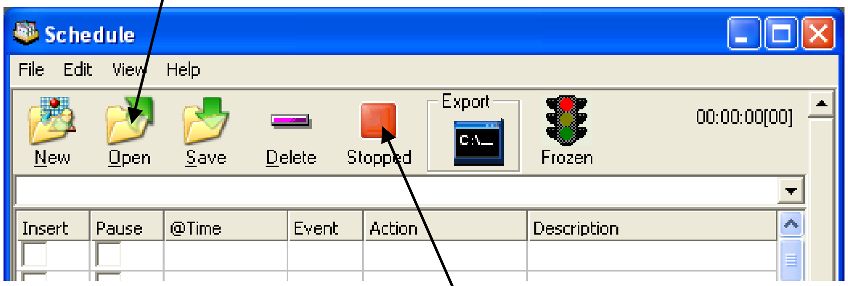
2. Startup the simulator using Simulator Instructor’s Job Aid section 7.3.

> *Note:*
> ***Prior to running the Schedule File, ensure no Event Files are Open.*** *If an existing Event File is Open prior to running the Schedule File, then any associated Event Files will not automatically load.*

3. Open Schedule.exe and Director.exe by clicking on the Icon in the Thunder Bar.

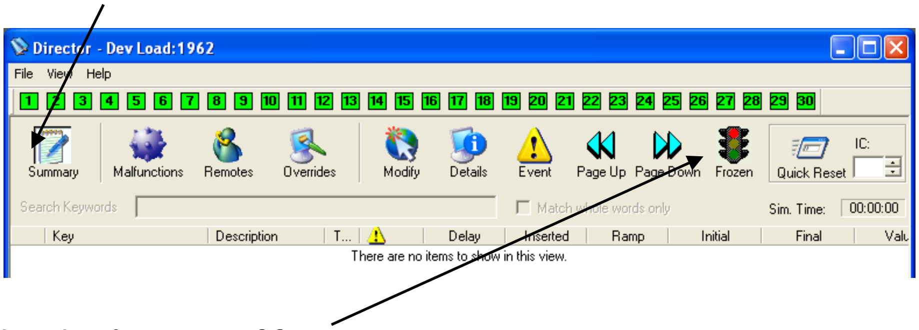
4. Set the Simulator to **IC-103** and perform switch check (Using Quick Reset in Director).

5. Click on “**Open”** in the Schedule window and Open Schedule File “**12-2017 NRC Exam Scenario 4.sch”** (in the Schedule Directory)



6. In Schedule window, click on the **“Stopped**” red block. The red block will change to a green arrow and indicate the scenario is active (“**Running**”).

7. Click the Summary tab in the Director window. Verify the schedule files are loaded and opened per Section B below. (Note: Any actions in the schedule file without a specific time will not load into the director until triggered.)



8. Take the simulator out of freeze.

9. Log on to all simulator PDS and SPDS computers.

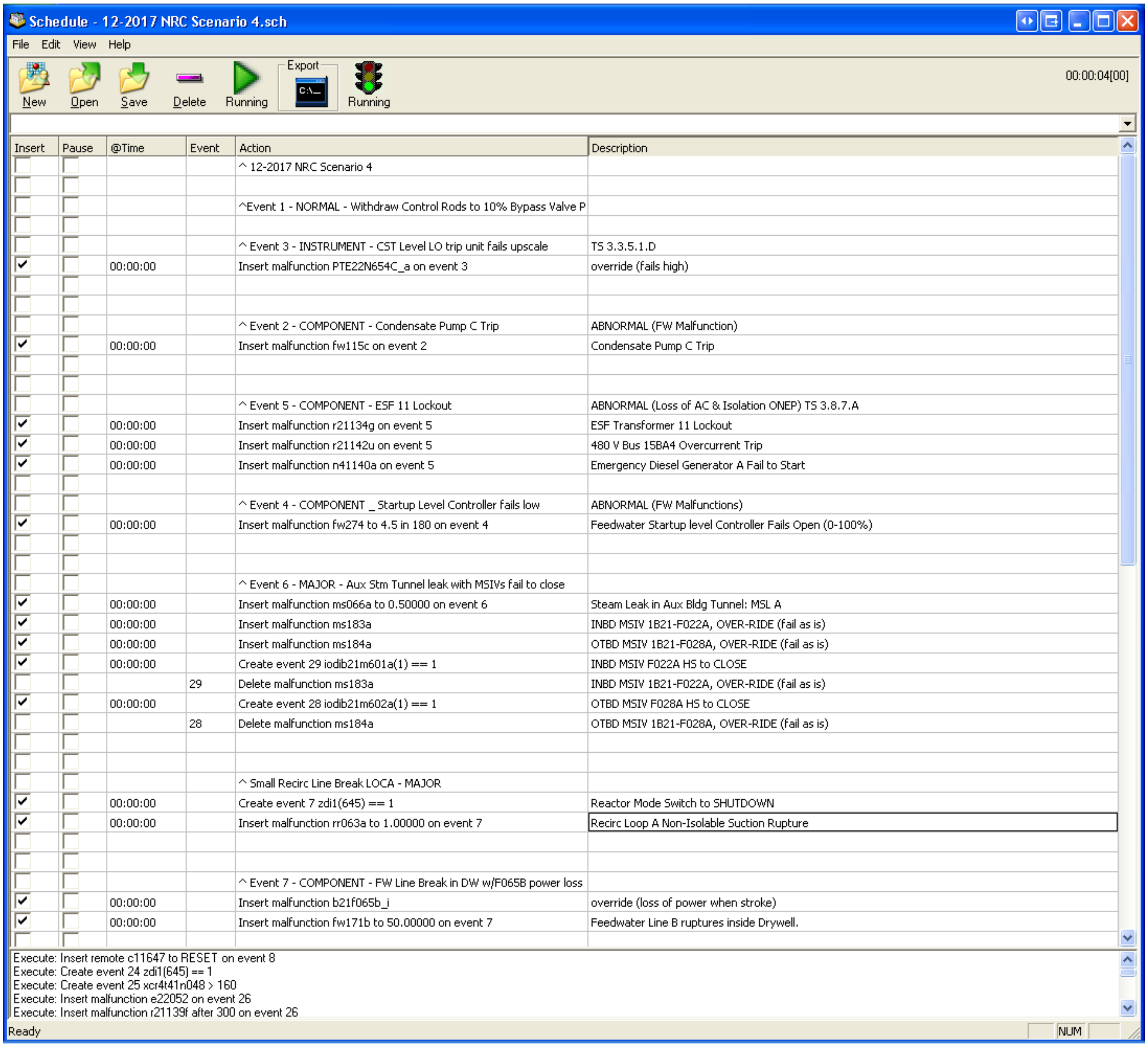
10. Verify or perform the following:

- IC-103
- Ensure the correct rod movement sequence available at the P680 and marked up through Step 96 complete
- Provide applicants with copy of 02-S-01-27, Operations Philosophy, Attachment 1, Control Rod Movement Expectation
- Provide applicants with copy of 04-1-01-C11-2, Rod Control and Information System, Section 3, Precautions and Limitations, and Section 4.3, RC&IS Rod/Rod Gang Notch Out.
- Ensure all procedures are marked as indicated for turnover conditions
- Set IRM recorder scales
- Advance all chart recorders and ensure all pens inking properly
- Clear any graphs and trends off of SPDS
- Place a tape flag on annunciator P680-4A2-C5, CONT ROD WITHDRAWL BLOCK
- Place a tape flag on annunciator P680-4A1-A7, CRD DRIVE WTR TO RX ΔP HI
- Place red tag on Generator Disconnect pushbutton

11. Run through any alarms and ensure alarms are on. **(Note: On T-Rex, to verify alarms are ON, the indicator will indicate "Alarms On").**

12. Place the simulator in Freeze.

File loaded verification:



Schedule - 12-2017 NRC Scenario 4.sch

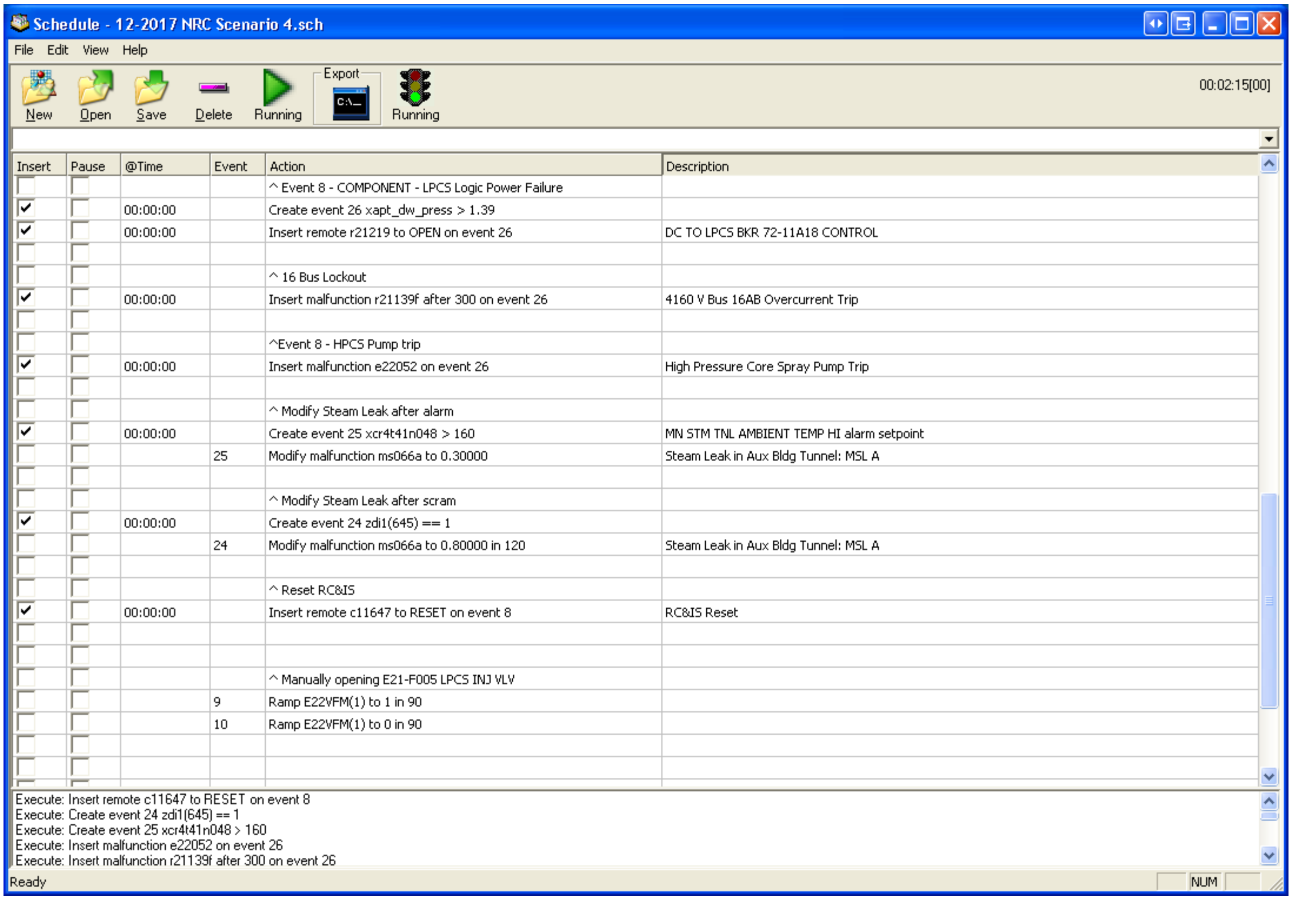
File Edit View Help

New Open Save Delete Running Export Running 00:00:04[00]

Insert	Pause	@Time	Event	Action	Description
				^ 12-2017 NRC Scenario 4	
				^Event 1 - NORMAL - Withdraw Control Rods to 10% Bypass Valve P	
				^ Event 3 - INSTRUMENT - CST Level LO trip unit fails upscale	TS 3.3.5.1.D
✓		00:00:00		Insert malfunction PTE22N654C_a on event 3	override (fails high)
				^ Event 2 - COMPONENT - Condensate Pump C Trip	ABNORMAL (FW Malfunction)
✓		00:00:00		Insert malfunction fw115c on event 2	Condensate Pump C Trip
				^ Event 5 - COMPONENT - ESF 11 Lockout	ABNORMAL (Loss of AC & Isolation ONEP) TS 3.8.7.A
✓		00:00:00		Insert malfunction r21134g on event 5	ESF Transformer 11 Lockout
✓		00:00:00		Insert malfunction r21142u on event 5	480 V Bus 15BA4 Overcurrent Trip
✓		00:00:00		Insert malfunction n41140a on event 5	Emergency Diesel Generator A Fail to Start
				^ Event 4 - COMPONENT _ Startup Level Controller fails low	ABNORMAL (FW Malfunctions)
✓		00:00:00		Insert malfunction fw274 to 4.5 in 180 on event 4	Feedwater Startup level Controller Fails Open (0-100%)
				^ Event 6 - MAJOR - Aux Stm Tunnel leak with MSIVs fail to close	
✓		00:00:00		Insert malfunction ms066a to 0.50000 on event 6	Steam Leak in Aux Bldg Tunnel: MSL A
✓		00:00:00		Insert malfunction ms183a	INBD MSIV 1B21-F022A, OVER-RIDE (fail as is)
✓		00:00:00		Insert malfunction ms184a	OTBD MSIV 1B21-F028A, OVER-RIDE (fail as is)
✓		00:00:00		Create event 29 iodib21m601a(1) == 1	INBD MSIV F022A HS to CLOSE
			29	Delete malfunction ms183a	INBD MSIV 1B21-F022A, OVER-RIDE (fail as is)
✓		00:00:00		Create event 28 iodib21m602a(1) == 1	OTBD MSIV F028A HS to CLOSE
			28	Delete malfunction ms184a	OTBD MSIV 1B21-F028A, OVER-RIDE (fail as is)
				^ Small Recirc Line Break LOCA - MAJOR	
✓		00:00:00		Create event 7 zdi1(645) == 1	Reactor Mode Switch to SHUTDOWN
✓		00:00:00		Insert malfunction rr063a to 1.00000 on event 7	Recirc Loop A Non-Isolable Suction Rupture
				^ Event 7 - COMPONENT - FW Line Break in DW w/F065B power loss	
✓		00:00:00		Insert malfunction b21f065b_i	override (loss of power when stroke)
✓		00:00:00		Insert malfunction fw171b to 50.00000 on event 7	Feedwater Line B ruptures inside Drywell.

Execute: Insert remote c11647 to RESET on event 8
Execute: Create event 24 zdi1(645) == 1
Execute: Create event 25 xcr4t41n048 > 160
Execute: Insert malfunction e22052 on event 26
Execute: Insert malfunction r21139f after 300 on event 26

Ready NUM



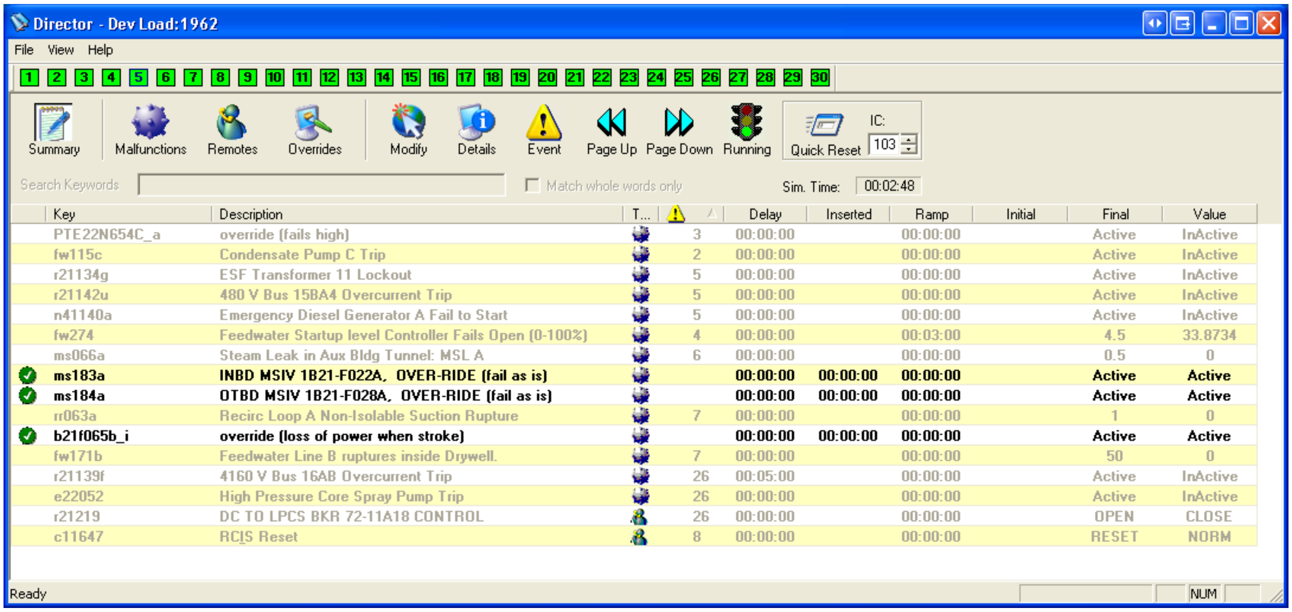
Schedule - 12-2017 NRC Scenario 4.sch

00:02:15[00]

Insert	Pause	@Time	Event	Action	Description
				^ Event 8 - COMPONENT - LPCS Logic Power Failure	
✔		00:00:00		Create event 26 xapt_dw_press > 1.39	
✔		00:00:00		Insert remote r21219 to OPEN on event 26	DC TO LPCS BKR 72-11A18 CONTROL
				^ 16 Bus Lockout	
✔		00:00:00		Insert malfunction r21139f after 300 on event 26	4160 V Bus 16AB Overcurrent Trip
				^Event 8 - HPCS Pump trip	
✔		00:00:00		Insert malfunction e22052 on event 26	High Pressure Core Spray Pump Trip
				^ Modify Steam Leak after alarm	
✔		00:00:00		Create event 25 xcr4t41n048 > 160	MN STM TNL AMBIENT TEMP HI alarm setpoint
			25	Modify malfunction ms066a to 0.30000	Steam Leak in Aux Bldg Tunnel: MSL A
				^ Modify Steam Leak after scram	
✔		00:00:00		Create event 24 zdi1(645) == 1	
			24	Modify malfunction ms066a to 0.80000 in 120	Steam Leak in Aux Bldg Tunnel: MSL A
				^ Reset RC&IS	
✔		00:00:00		Insert remote c11647 to RESET on event 8	RC&IS Reset
				^ Manually opening E21-F005 LPCS INJ VLV	
			9	Ramp E22VFM(1) to 1 in 90	
			10	Ramp E22VFM(1) to 0 in 90	

Execute: Insert remote c11647 to RESET on event 8
Execute: Create event 24 zdi1(645) == 1
Execute: Create event 25 xcr4t41n048 > 160
Execute: Insert malfunction e22052 on event 26
Execute: Insert malfunction r21139f after 300 on event 26

Ready NUM



Director - Dev Load:1962

IC: 103 Sim. Time: 00:02:48

Key	Description	T...	⚠	Delay	Inserted	Ramp	Initial	Final	Value
PTE22N654C_a	override (fails high)		3	00:00:00		00:00:00		Active	InActive
fw115c	Condensate Pump C Trip		2	00:00:00		00:00:00		Active	InActive
r21134g	ESF Transformer 11 Lockout		5	00:00:00		00:00:00		Active	InActive
r21142u	480 V Bus 15BA4 Overcurrent Trip		5	00:00:00		00:00:00		Active	InActive
n41140a	Emergency Diesel Generator A Fail to Start		5	00:00:00		00:00:00		Active	InActive
fw274	Feedwater Startup level Controller Fails Open (0-100%)		4	00:00:00		00:03:00		4.5	33.8734
ms066a	Steam Leak in Aux Bldg Tunnel: MSL A		6	00:00:00		00:00:00		0.5	0
ms183a	INBD MSIV 1B21-F022A, OVER-RIDE (fail as is)			00:00:00	00:00:00	00:00:00		Active	Active
ms184a	OTBD MSIV 1B21-F028A, OVER-RIDE (fail as is)			00:00:00	00:00:00	00:00:00		Active	Active
rr063a	Recirc Loop A Non-Isolable Suction Rupture		7	00:00:00		00:00:00		1	0
b21f065b_i	override (loss of power when stroke)			00:00:00	00:00:00	00:00:00		Active	Active
fw171b	Feedwater Line B ruptures inside Drywell.		7	00:00:00		00:00:00		50	0
r21139f	4160 V Bus 16AB Overcurrent Trip		26	00:05:00		00:00:00		Active	InActive
e22052	High Pressure Core Spray Pump Trip		26	00:00:00		00:00:00		Active	InActive
r21219	DC TO LPCS BKR 72-11A18 CONTROL		26	00:00:00		00:00:00		OPEN	CLOSE
c11647	RCIS Reset		8	00:00:00		00:00:00		RESET	NORM

Ready NUM

Procedures that may be used in this scenario:

Procedure No.	Rev	Procedure Title
03-1-01-IOI-1	172	Cold Shutdown To Generator Carrying Minimum Load
04-1-01-C11-1	154	Control Rod Drive Hydraulic System
04-1-01-C41-1	123	Standby Liquid Control System
04-1-01-E12-1	147	Residual Heat Removal System
04-1-01-N21-1	74	Feedwater System
04-1-01-N32-2	33	Turbine Generator Control
04-1-01-R21-17	10	ESF Bus 17AC
04-1-02-1H13-P601	161	Alarm Response Instruction Panel No.: 1H13-P601
04-1-02-1H13-P680	233	Alarm Response Instruction Panel No.:1H13-P680
04-1-02-1H13-P864	31	Alarm Response Instruction Panel No.: 1H13-P864
04-1-02-1H13-P870	154	Alarm Response Instruction Panel No.: 1H13-P870
04-S-02-SH13-P807	32	Alarm Response Instruction Panel No.: SH13-P807
05-1-02-I-1	130	Reactor Scram
05-1-02-I-2	37	Turbine and Generator Trips
05-1-02-I-4	51	Loss of AC Power
05-1-02-III-3	115	Reduction In Recirculation System Flow Rate
05-1-02-III-5	49	Automatic Isolations
05-1-02-IV-1	117	Control Rod / Drive Malfunctions
05-S-01-EP-1	36	Emergency / Severe Accident Procedure Support Documents
05-S-01-EP-2	45	RPV Control
Tech Spec 3.1.3		
Tech Spec 3.8.1		

Expected Alarms:

P680-4A2-C5, CONT ROD WITHDRAWL BLOCK
P680-1A-A3, CNDS PMP C TRIP
P601-16A-H5, HPCS SYS OOSVC
P680-3A-A3, RX LVL 40"/32" HI/LO
P680-2A-C9, DFCS TROUBLE
P807-4A-E6, ESF XFMR 11 TROUBLE
P807-4A-B2, ESF XFMR 11 LOCKOUT TRIP
P807-1A-B5, ESF DIST BUSES INCM FDR 152-1902
P807-1A-B4, ESF DIST BUSES INCM FDR 152-1901
P864-1A-A3, 4.16KV BUS 15AA UNDERVOLTAGE
P864-1A-D1, DG 11 AUTO START NOT AVAIL
P864-1A-D2, DIV 1 DSL GEN TROUBLE
P864-1A-D3, 480V LCC 15BA1 UNDERVOLT
P864-1A-D4, 480V LCC 15BA2 UNDERVOLT
P864-1A-F4, 480V LCC 15BA6 UNDERVOLT
P864-1A-F3, 480V LCC 15BA5 UNDERVOLT
P864-1A-E4, 480V LCC 15BA4 UNDERVOLT
P864-1A-E3, 480V LCC 15BA3 UNDERVOLT
P680-5A-C4, SCRAM PILOT VLV AIR HDR PRESS LO
P680-4A2-A4, RC&IS INOP
P864-1A-A4, 4.16KV BUS 15AA INCM FDRS TRIP
P807-3A-H1, STATIC INVRTR 1Y80 TROUBLE
P807-3A-G4, STATIC INVRTR 1Y79 TROUBLE
P807-3A-G3, STATIC INVRTR 1Y98 TROUBLE
P807-1A-A6, ESF XFMR 11 INCM FDR 552-1104 TRIP
P601-18A-A3, MSL PIPE TNL CH-B TEMP HI
P601-18A-A4, MSL PIPE TNL CH-C TEMP HI
P601-19A-A3, MSL PIPE TNL CH-A TEMP HI
P601-19A-A4, MSL PIPE TNL CH-D TEMP HI

INITIAL CONDITIONS

A. Plant Status:

- Reactor startup is in progress with power at approximately 4%:
  - Step 45 of IOI 03-1-01-1, Attachment XV
  - Step 96 of Control Rod Movement Sequence is complete
  - SJAE B is in service

- Condensate System is lined up as follows:
  - Condensate Pumps A and C in service
  - Condensate Booster Pump C in service
  - Reactor Feed Pump A in service at approximately 950 psig discharge pressure
  - CFFF is in service
  - 4 Deep-Bed Condensate Demineralizers are in service

- Annunciators P680-4A2-C5, CONT ROD WITHDRAWAL BLOCK, and P680-4A1-A7, CRD DRIVE WTR TO RX ΔP HI, are flagged as expected annunciators

B. Tech. Spec. Limitations in effect: None

C. Significant problems/abnormalities: None

D. Integrated Risk: High

E. Division Work Week: Division 2

F. Evolutions/maintenance for the up-coming shift :

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