GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE EMERGENCY & ABNORMAL PLANT EVOLUTIONS - TIER 1 GROUP 1 (RO/SRO)								Form ES-401-1
E/APE #/NAME/SAFETY FUNCTION	**K1**	**K2**	**K3**	**A1**	**A2**	**G**	**TOPIC(S)**	**IR**	**#**
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4 CFR						2.4.4	Given plant conditions, parameters, and a loss of the recirculation system, determine appropriate actions.	4.0	1 833
295003 Partial or Complete Loss of AC Power/ 6 CFR			01					3.3	2 464
295004 Partial or Complete Loss of DC Power / 6 CFR				02			Given plant conditions and a loss of DC power, determine the effect to the *SDC* system.	3.8	3 834
295005 Main Turbine Generator Trip / 3 CFR	03						Following a reactor scram and subsequent main turbine generator trip, determine the effects of manual bypass valve operation on reactor water level.	3.5	4 835
295006 SCRAM / 1 CFR	02						Given plant conditions following a reactor scram, determine if adequate shutdown margin exists.	3.4	5 836
295016 Control Room Abandonment / 7 CFR				01			Describe the method used to manually scram the reactor after the control room has been abandoned.	3.8	6 837
295018 Partial or Complete Loss of CCW / 8 CFR	01						Given plant conditions and a partial loss of Component Cooling Water, determine the necessary actions to ensure the plant remains/returns to a safe condition.	3.5	7 838
295019 Partial or Complete Loss of Inst. Air / 8 CFR					01		Given indications of a partial loss of Instrument Air determine a method to restore Instrument Air system pressure.	3.5	8 548
295021 Loss of Shutdown Cooling / 4 CFR			01				Given specific plant conditions following a loss of Shutdown Cooling, determine the reason for raising reactor water level.	3.3	9 078a
295023 Refueling Accidents / 8 CFR	03						Determine the correct operator response to inadvertent criticality following a refueling accident.	3.7	10 848
295024 High Drywell Pressure / 5 CFR						2.1.23	Given plant conditions and high drywell pressure, determine the method to lower drywell pressure.	3.9	11 849
295025 High Reactor Pressure / 3 CFR			*06*					4.2	12 690
295026 Suppression Pool High Water Temp. / 5 CFR			01				Given an ATWS condition, describe the EP bases for lowering reactor pressure as Suppression Pool temperature rises.	3.8	13 840
295027 High Containment Temperature / 5 CFR				03			Given rising Containment temperature, describe the necessary actions to maintain the plant/containment in a safe condition.	3.5	14 844
PAGE 1 TOTAL TIER 1 GROUP 1	4	0	4	3	1	2	PAGE TOTAL # QUESTIONS	14	

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE EMERGENCY & ABNORMAL PLANT EVOLUTIONS - TIER 1 GROUP 1 (RO/SRO)	Form ES-401-1

E/APE #/NAME/SAFETY FUNCTION	K1	K2	K3	A1	A2	G	TOPIC(S)	IMP	#
295028 High Drywell Temperature / 5 CFR	02						Given plant conditions and elevated drywell temperature, determine the effects to control room reactor water level indication.	2.9	15 845
295030 Low Suppression Pool Water Level / 5 CFR						2. 2. 12	Given a low suppression pool level condition, determine the effects to other plant systems.	3.0	16 846
295031 Reactor Low Water Level / 2 CFR				04			Given plant conditions, describe the operation of the High Pressure Core Spray system following a LOCA.	4.3	17 847
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1 CFR			06				Given plant conditions and an ATWS condition, determine the availability of the main condenser as a heat sink.	3.8	18 850
295038 High Offsite Release Rate / 9 CFR					01		Given a radioactive release from the plant, determine when it is considered to be offsite.	3.3	19 851
600000 Plant Fire On Site / 8			04				Determine the required procedural actions for a fire on the plant site.	2.8	20 852
PAGE 2 TOTAL TIER 1 GROUP 1	1	0	2	1	1	1	PAGE TOTAL # QUESTIONS	6	
PAGE 1 TOTAL TIER 1 GROUP 1	4	0	4	3	1	2	PAGE TOTAL # QUESTIONS	14	
TIER 1 GROUP 1 TOTALS	5	0	6	4	2	3		20	

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE EMERGENCY & ABNORMAL PLANT EVOLUTIONS - TIER 1 GROUP 2 **(RO/SRO)**								
E/APE #/NAME/SAFETY FUNCTION	**K1**	**K2**	**K3**	**A1**	**A2**	**G**	**TOPIC(S)**	**IMP**	**#**
295002 Loss of Main Condenser Vacuum / 3 CFR		01					Given plant conditions and degrading main condenser vacuum, determine the automatic plant response (RPS actuation).	3.5	21 854
295007 High Reactor Pressure / 3 CFR						2. 4. 35	Determine the conditions necessary to require connection of an alternate air source to the SRVs.	3.3	22 855
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2									
295010 High Drywell Pressure / 5									
295011 High Containment Temperature / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Water Temp. / 5 CFR					02		Describe the preferred method to minimize localized suppression pool heating when using the SRVs to control reactor pressure without suppression cooling in service.	3.2	23 856
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1									
295017 High Offsite Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1									
295029 High Suppression Pool Water Level / 5									
PAGE 1 TOTAL TIER 1 GROUP 2	0	1	0	0	1	1	PAGE TOTAL # QUESTIONS	3	

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE EMERGENCY & ABNORMAL PLANT EVOLUTIONS - TIER 1 GROUP 2 (RO/SRO)								Form ES-401-1
E/APE #/NAME/SAFETY FUNCTION	**K1**	**K2**	**K3**	**A1**	**A2**	**G**	**TOPIC(S)**	**IMP**	**#**
295032 High Secondary Containment Area Temperature / 5 CFR			02				Given plant conditions including elevated Auxiliary Building temperatures, describe the conditions that would require a reactor scram.	3.5	24 857
295033 High Secondary Containment Area Radiation Levels / 9									
295034 Secondary Containment Ventilation High Radiation / 9 CFR		03					Given plant conditions including elevated Auxiliary Building radiation levels, describe the conditions that would automatically start the Standby Gas Treatment system.	4.3	25 858
295035 Secondary Containment High Differential Pressure / 5 CFR	02						Given accident conditions and a Standby Gas Treatment system failure, determine the type of release.		26 859
295036 Secondary Containment High Sump/Area Water Level / 5 CFR				01			Describe the system logic used by the Auxiliary Building Floor Drain system to contain a significant CCW system rupture.	3.2	27 860
500000 High CTMT Hydrogen Conc. / 5									
PAGE 2 TOTAL TIER 1 GROUP 2	1	1	1	1	0	0	PAGE TOTAL # QUESTIONS	4	
PAGE 1 TOTAL TIER 1 GROUP 2	0	1	0	0	1	1	PAGE TOTAL # QUESTIONS	3	
TIER 1 GROUP 2 TOTALS	1	2	1	1	1	1		7	

PLANT SYSTEMS - TIER 2 GROUP 1 (RO/SRO)

SYSTEM #/NAME	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
203000 RHR/LPCI: Injection Mode CFR					02							Given plant conditions, describe the design features and limits of the RHR pump manual override feature.	3.5	28 861
205000 Shutdown Cooling CFR				04								Describe the RHR Shutdown Cooling system NPSH interlocks.	2.6	29 862
206000 HPCI												N/A GGNS		
207000 Isolation (Emergency) Condenser												N/A GGNS		
209001 LPCS CFR										01		Given degraded plant conditions during a LOCA, describe LPCS manual operation.	3.8	30 863
209002 HPCS CFR										09		Describe available methods to raise/lower suppression pool level using HPCS.	3.4	31 864
209002 HPCS CFR											2.1.28	Describe the bases for the HPCS injection valve high reactor water level interlock.	3.2	32 865
211000 SLC CFR								02				Predict the SLC system indication and response with indication the squib valve failed to actuate and follow up actions.	3.6	33 866
212000 RPS CFR								12				Given plant conditions including a partial main turbine stop/control valve closure, determine the effect to RPS.	4.0	34 867
215003 IRM CFR					03							Describe the reason for the precaution concerning driving IRMs during surveillance activities.	3.0	35 868
215004 Source Range Monitor CFR											2.2.33	Describe the SRM precaution warning of a potential control rod block even if the channel is bypassed.	2.5	36 869
215005 APRM / LPRM CFR		02										Given a partial loss of plant electrical power, determine the effect to the APRMs.	2.6	37 870
217000 RCIC CFR							02					Predict how a reactor pressure change will affect RCIC system flow.	3.3	38 871
PAGE 1 TOTAL												PAGE TOTAL #		

TIER 2 GROUP 1	0	1	0	1	2	0	1	2	0	2	2	QUESTIONS	11

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE PLANT SYSTEMS - TIER 2 GROUP 1 (RO/SRO)	Form ES-401-1

SYSTEM #/NAME	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	TOPIC(S)	IMP	#
218000 ADS CFR		01										Describe the relationship between ADS Logic power and the operation of the ADS logic.	3.1	39 872
223002 PCIS / Nuclear Steam Supply Shutoff CFR								03				Determine the operator actions required to mitigate a NSSSS logic failure.	3.0	40 873
239002 SRVs CFR				09								Describe the design features available to determine if a SRV is open.	3.7	41 874
259002 Reactor Water Level Control CFR								04				*Describe the operator response to a failure of RFPT speed control with speed rising.*	3.0	42 875
259002 Reactor Water Level Control CFR										06		Describe prerequisites for transferring the Feedwater system to 3-element control.	3.1	43 233 a
261000 SGTS CFR									03			Describe the SGTS damper logic following system initiation.	3.0	44 876
262001 AC Electrical Distribution CFR						01						Given plant conditions and a partial loss of DC power, determine the affect to the AC distribution system.	3.1	45 877
262002 UPS (AC/DC) CFR				01								Given plant conditions and degraded AC power, determine the status of plant inverters.	3.1	46 878
263000 DC Electrical Distribution CFR				01								Given a loss of AC power to battery chargers, determine the affects to the DC distribution system.		47 879
264000 EDGs CFR								10				Describe EDG response to a LOCA.	3.9	48 880
264000 EDGs CFR											2.4.48	Determine EDG status from control room alarms and indications and any required operator actions to improve plant conditions.	3.5	49 881
300000 Instrument Air CFR			01									Determine the effect on the plant given a loss of Instrument Air to the containment.	2.7	50 882
PAGE 2 TOTAL TIER 2 GROUP 1	0	1	1	3	0	1	0	3	1	1	1	PAGE TOTAL # QUESTIONS	12	

PLANT SYSTEMS - TIER 2 GROUP 1 (RO/SRO)

SYSTEM #/NAME	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
300000 Instrument Air CFR						13						Determine the affect of a clogged filter on the Instrument Air system.	2.8	51 883
400000 Component Cooling Water CFR	04											Determine the method used to confirm a reactor coolant leak into the CCW system.	2.9	52 884
400000 Component Cooling Water CFR							02					Determine the affect to the plant if the CCW temperature control fails.	2.8	53 885
PAGE 3 TOTAL TIER 2 GROUP 1	1	0	0	0	0	1	1	0	0	0	0	PAGE TOTAL # QUESTIONS	3	
PAGE 1 TOTAL TIER 2 GROUP 1	0	1	0	1	2	0	1	2	0	2	2	PAGE TOTAL # QUESTIONS	11	
PAGE 2 TOTAL TIER 2 GROUP 1	0	1	1	3	0	1	0	3	1	1	1	PAGE TOTAL # QUESTIONS	12	
TIER 2GROUP 1 TOTALS	1	2	1	4	2	2	2	5	1	3	3		26	

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE PLANT SYSTEMS - TIER 2 GROUP 2 (RO/SRO)	Form ES-401-1

SYSTEM #/NAME	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
201001 CRD Hydraulic CFR														
201002 RMCS												N/A GGNS		
201003 Control Rod and Drive Mechanism CFR														
201004 RSCS												N/A GGNS		
201005 RCIS CFR					10							Describe the purpose for the rod withdrawal limiter.	3.2	54 886
201006 RWM												N/A GGNS		
202001 Recirculation CFR											*2.4.11*		*3.4*	55 887
202002 Recirculation Flow Control CFR41.6	01											Given plant conditions, determine any automatic actions associated with the Recirculation System HPUs.	3.5	56 888
204000 RWCU CFR				06								Determine the correct flow path to use RWCU as an alternate shutdown cooling.	2.6	57 889
214000 RPIS												N/A GGNS		
215001 Traversing In-Core Probe CFR														
215002 RBM												N/A GGNS		
PAGE 1 TOTAL TIER 2 GROUP 2	1	0	0	1	1	0	0	0	0	0	1	PAGE TOTAL # QUESTIONS	4	

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE PLANT SYSTEMS - TIER 2 GROUP 2 (RO/SRO)	Form ES-401-1

SYSTEM #/NAME	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
216000 Nuclear Boiler Instrumentation CFR														
219000 RHR /LPCI Suppression Pool Cooling Mode CFR														
223001 Primary CTMT and Auxiliaries CFR	08											Determine the limitations to SRV usage given a reduced suppression pool level.	3.6	58 890
226001 RHR/LPCI: CTMT Spray Mode CFR														
230000 RHR/LPCI: Torus/Pool Spray Mode												N/A GGNS		
233000 Fuel Pool Cooling and Cleanup CFR														
234000 Fuel Handling Equipment CFR														
239001 Main and Reheat Steam CFR			04									Given plant conditions including a MSIV closure, determine the affect to the Offgas system.	2.8	59 891
239003 MSIV Leakage Control CFR	02											Explain the relationship between the MSIV Leakage Control system and SGTS.	2.9	60 892
241000 Reactor/Turbine Pressure Regulator CFR											2.4.6	Describe the bases for each of the Scram ONEP immediate actions.	3.1	61 893
PAGE 2 TOTAL TIER 2 GROUP 2	2	0	1	0	0	0	0	0	0	0	1	PAGE TOTAL # QUESTIONS	4	

GRAND GULF NUCLEAR STATION

BWR EXAMINATION OUTLINE
PLANT SYSTEMS - TIER 2 GROUP 2 (RO/SRO)

Form ES-401-1

SYSTEM #/NAME	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	TOPIC(S)	IMP	#
245000 Main Turbine Gen./Aux. CFR									02			Determine main turbine critical speeds as it is rolled to rated speed.	2.8	62 894
256000 Reactor Condensate CFR														
259001 Reactor Feedwater CFR								03				Determine necessary actions and priorities immediately after a single condensate pump trips with the plant at rated conditions.	3.6	63 895
268000 Radwaste CFR	04											Determine the Drywell Floor Drains indications available to detect drywell general area leakage.	2.7	64 896
271000 Offgas CFR														
272000 Radiation Monitoring CFR														
286000 Fire Protection CFR														
288000 Plant Ventilation CFR														
290001 Secondary CTMT CFR				03								Determine inputs to the Fuel Pool leak detection standpipe.	2.8	65 897
290003 Control Room HVAC CFR														
290002 Reactor Vessel Internals CFR														
PAGE 3 TOTAL TIER 2 GROUP 2	1	0	0	1	0	0	0	1	1	0	0	PAGE TOTAL # QUESTIONS	4	
PAGE 1 TOTAL TIER 2 GROUP 2	1	0	0	1	1	0	0	0	0	0	1	PAGE TOTAL # QUESTIONS	4	
PAGE 2 TOTAL TIER 2 GROUP 2	2	0	1	0	0	0	0	0	0	0	1	PAGE TOTAL # QUESTIONS	4	
TIER 2 GROUP 2 TOTALS	4	0	1	2	1	0	0	1	1	0	2		12	

Facility: **Grand Gulf Nuclear Station** Date of Exam: **12 August 2005**

Category	K/ A#	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct Of Operations	2.1.19	Given plant conditions and the PDS computer, determine necessary actions based on PBDS counts.	3.0	66 898		
	2.1.25	Given plant conditions and EOP-3 graphs, determine the correct mitigation strategy.	2.8	67 899		
	2.1.29	Determine the correct locking device color coding for locked components.	3.4	68 237a		
	2.1					
	2.1					
	2.1					
	Subtotal			3		
2. Equipment Control	2.2.1	Given plant conditions, determine proper operation of the IRMs.	3.7	69 900		
	2.2.30	Discuss the duties of the operator assigned to communicate with the refueling floor SRO during core alterations.	3.5	70 901		
	2.2					
	2.2					
	2.2					
	2.2					
	Subtotal			2		
3. Radiation Control	2.3.1	Given the need to enter a high radiation area, determine the allowed time in the area to prevent exceeding the administrative exposure limits.	2.6	71 902		
	2.3.4	Given plant conditions and applicable Emergency Planning Procedures, determine the radiation exposure limits that are in effect.	2.5	72 903		
	2.3					
	2.3					
	2.3					
	2.3					
	Subtotal			2		
4. Emergency Procedures / Plan	2.4.20	Given plant conditions, determine the bases for any applicable EOP cautions.	3.3	73 904		
	2.4.25	Given plant conditions including a fire, determine the proper response.	2.9	74 905		
	2.4.43	Given plant conditions and Emergency Plan Procedures, determine the available emergency communications systems.	2.8	75 906		
	2.4					
	2.4					
	2.4					
	Subtotal			3		
Tier 3 Point Total				10		7

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE EMERGENCY & ABNORMAL PLANT EVOLUTIONS - TIER 1 GROUP 1 (RO/SRO)						Form ES-401-1		
E/APE #/NAME/SAFETY FUNCTION	**K1**	**K2**	**K3**	**A1**	**A2**	**G**	**TOPIC(S)**	**IR**	**#**
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4 CFR						2. 4. 4	Given plant conditions, parameters, and a loss of the recirculation system, determine appropriate actions.	4.3	1 833
295003 Partial or Complete Loss of AC Power/ 6 CFR			01					3.5	2 464
295004 Partial or Complete Loss of DC Power / 6 CFR				02			Given plant conditions and a loss of DC power, determine the effect to the SDC system.	4.1	3 834
295005 Main Turbine Generator Trip / 3 CFR	03						Following a reactor scram and subsequent main turbine generator trip, determine the effects of manual bypass valve operation on reactor water level.	3.7	4 835
295006 SCRAM / 1 CFR	02						Given plant conditions following a reactor scram, determine if adequate shutdown margin exists.	3.7	5 836
295016 Control Room Abandonment / 7 CFR				01			Describe the method used to manually scram the reactor after the control room has been abandoned.	3.9	6 837
295018 Partial or Complete Loss of CCW / 8 CFR	01						Given plant conditions and a partial loss of Component Cooling Water, determine the necessary actions to ensure the plant remains/returns to a safe condition.	3.6	7 838
295019 Partial or Complete Loss of Inst. Air / 8 CFR					01		Given indications of a partial loss of Instrument Air determine a method to restore Instrument Air system pressure.	3.6	8 548
295021 Loss of Shutdown Cooling / 4 CFR			01				Given specific plant conditions following a loss of Shutdown Cooling, determine the reason for raising reactor water level.	3.4	9 078 a
295023 Refueling Accidents / 8 CFR	03						Determine the correct operator response to inadvertent criticality following a refueling accident.	4.0	10 848
295024 High Drywell Pressure / 5 CFR						2. 1. 23	Given plant conditions and high drywell pressure, determine the method to lower drywell pressure.	4.0	11 849
295025 High Reactor Pressure / 3 CFR			*06*					*4.4*	12 690
295026 Suppression Pool High Water Temp. / 5 CFR			01				Given an ATWS condition, describe the EP bases for lowering reactor pressure as Suppression Pool temperature rises.	4.1	13 840
295027 High Containment Temperature / 5 CFR				03			Given rising Containment temperature, describe the necessary actions to maintain the plant/containment in a safe condition.	3.8	14 844
PAGE 1 TOTAL TIER 1 GROUP 1	4	0	4	3	1	2	PAGE TOTAL # QUESTIONS	14	

E/APE #/NAME/SAFETY FUNCTION	K1	K2	K3	A1	A2	G	TOPIC(S)	IMP	#
295028 High Drywell Temperature / 5 CFR	02						Given plant conditions and elevated drywell temperature, determine the effects to control room reactor water level indication.	3.1	15 845
295030 Low Suppression Pool Water Level / 5 CFR						2.2.12	Given a low suppression pool level condition, determine the effects to other plant systems.	3.4	16 846
295031 Reactor Low Water Level / 2 CFR				04			Given plant conditions, describe the operation of the High Pressure Core Spray system following a LOCA.	4.2	17 847
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1 CFR			06				Given plant conditions and an ATWS condition, determine the availability of the main condenser as a heat sink.	4.1	18 850
295038 High Offsite Release Rate / 9 CFR					01		Given a radioactive release from the plant, determine when it is considered to be offsite.	4.3	19 851
600000 Plant Fire On Site / 8			04				Determine the required procedural actions for a fire on the plant site.	3.4	20 852
295004 Partial or Complete Loss of DC Power / 6 CFR					02		Given a loss of Division 1 DC logic power, determine the affect to the Division 1 ECCS.	3.9*	76 908
295005 Main Turbine Generator Trip / 3 CFR						2.3.5	Given plant data including current area dose rates, determine the required personnel monitoring equipment needed to enter the main turbine/generator area to investigate the cause for a trip.	2.5*	77 909
295026 Suppression Pool High Water Temp. / 5 CFR					03		Given plant conditions including rising Suppression Pool temperature, interpret HCTL and determine appropriate actions.	4.0*	78 910
295027 High Containment Temperature / 5 CFR						2.2.22	Explain the bases for the Technical Specification Containment average air temperature limit.	4.1*	79 911
295030 Low Suppression Pool Water Level / 5 CFR					02		Given low suppression pool water level, determine if suppression pool temperature can/cannot be measured and why.	3.9*	80 912
295038 High Offsite Release Rate / 9 CFR						2.2.28	Given a severe case fuel handling accident, explain the processes designed to prevent high offsite release rates.	3.5*	81 913
600000 Plant Fire On Site / 8					16		*Describe the basis for separating vital equipment from the Main Control Room during a fire in the Main Control Room.*	3.5*	82 914
							* SRO Only Questions		
PAGE 2 TOTAL TIER 1 GROUP 1	1	0	2	1	5	4	PAGE TOTAL # QUESTIONS	13	
PAGE 1 TOTAL TIER 1 GROUP 1	4	0	4	3	1	2	PAGE TOTAL # QUESTIONS	14	
TIER 1 GROUP 1 TOTALS	5	0	6	4	6	6		27	

E/APE #/NAME/SAFETY FUNCTION	K1	K2	K3	A1	A2	G	TOPIC(S)	IMP	#
295002 Loss of Main Condenser Vacuum / 3 CFR		01					Given plant conditions and degrading main condenser vacuum, determine the automatic plant response (RPS actuation).	3.5	21 854
295007 High Reactor Pressure / 3 CFR						2.4.35	Determine the conditions necessary to require connection of an alternate air source to the SRVs.	3.5	22 855
295008 High Reactor Water Level / 2									
295009 Low Reactor Water Level / 2									
295010 High Drywell Pressure / 5									
295011 High Containment Temperature / 5									
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Water Temp. / 5 CFR					02		Describe the preferred method to minimize localized suppression pool heating when using the SRVs to control reactor pressure without suppression cooling in service.	3.5	23 856
295014 Inadvertent Reactivity Addition / 1									
295015 Incomplete SCRAM / 1									
295017 High Offsite Release Rate / 9									
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1									
295029 High Suppression Pool Water Level / 5									
PAGE 1 TOTAL TIER 1 GROUP 2	0	1	0	0	1	1	PAGE TOTAL # QUESTIONS	3	

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE EMERGENCY & ABNORMAL PLANT EVOLUTIONS - TIER 1 GROUP 2 (RO/SRO)							Form ES-401-1
E/APE #/NAME/SAFETY FUNCTION	**K1**	**K2**	**K3**	**A1**	**A2**	**G**	**TOPIC(S)**	**IMP**	**#**
295032 High Secondary Containment Area Temperature / 5 CFR			02				Given plant conditions including elevated Auxiliary Building temperatures, describe the conditions that would require a reactor scram.	3.8	24 857
295033 High Secondary Containment Area Radiation Levels / 9									
295034 Secondary Containment Ventilation High Radiation / 9 CFR		03					Given plant conditions including elevated Auxiliary Building radiation levels, describe the conditions that would automatically start the Standby Gas Treatment system.	4.5	25 858
295035 Secondary Containment High Differential Pressure / 5 CFR	02						Given accident conditions and a Standby Gas Treatment system failure, determine the type of release.	4.2	26 859
295036 Secondary Containment High Sump/Area Water Level / 5 CFR				01			Describe the system logic used by the Auxiliary Building Floor Drain system to contain a significant CCW system rupture.	3.3	27 860
500000 High CTMT Hydrogen Conc. / 5									
295011 High Containment Temperature / 5 CFR					01		Given LOCA conditions, determine when containment spray should be initiated.	3.9*	83 915
295014 Inadvertent Reactivity Addition / 1 CFR						2.1.14	Given a control rod drifting out with the plant at power, determine any necessary notifications.	3.3*	84 916
295020 Inadvertent Cont. Isolation / 5 & 7 CFR					03		Given a partial MSIV closure, determine the affect on reactor power.	3.7*	85 917
							* SRO Only Questions		
PAGE 2 TOTAL TIER 1 GROUP 2	1	1	1	1	2	1	PAGE TOTAL # QUESTIONS	7	
PAGE 1 TOTAL TIER 1 GROUP 2	0	1	0	0	1	1	PAGE TOTAL # QUESTIONS	3	
TIER 1 GROUP 2 TOTALS	1	2	1	1	3	2		10	

PLANT SYSTEMS - TIER 2 GROUP 1 (RO/SRO)

SYSTEM #/NAME	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
203000 RHR/LPCI: Injection Mode CFR					02							Given plant conditions, describe the design features and limits of the RHR pump manual override feature.	3.7	28 861
205000 Shutdown Cooling CFR				04								Describe the RHR Shutdown Cooling system NPSH interlocks.	2.6	29 862
206000 HPCI												N/A GGNS		
207000 Isolation (Emergency) Condenser												N/A GGNS		
209001 LPCS CFR										01		Given degraded plant conditions during a LOCA, describe LPCS manual operation.	3.6	30 863
209002 HPCS CFR										09		Describe available methods to raise/lower suppression pool level using HPCS.	3.5	31 864
209002 HPCS CFR											2.1.28	Describe the bases for the HPCS injection valve high reactor water level interlock.	3.3	32 865
211000 SLC CFR								02				Predict the SLC system indication and response with indication the squib valve failed to actuate and follow up actions.	3.9	33 866
212000 RPS CFR								12				Given plant conditions including a partial main turbine stop/control valve closure, determine the effect to RPS.	4.1	34 867
215003 IRM CFR					03							Describe the reason for the precaution concerning driving IRMs during surveillance activities.	3.1	35 868
215004 Source Range Monitor CFR											2.2.33	Describe the SRM precaution warning of a potential control rod block even if the channel is bypassed.	2.9	36 869
215005 APRM / LPRM CFR		02										Given a partial loss of plant electrical power, determine the effect to the APRMs.	2.8	37 870
217000 RCIC CFR							02					Predict how a reactor pressure change will affect RCIC system flow.	3.3	38 871
PAGE 1 TOTAL TIER 2 GROUP 1	0	1	0	1	2	0	1	2	0	2	2	PAGE TOTAL # QUESTIONS	11	

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE PLANT SYSTEMS - TIER 2 GROUP 1 (RO/SRO)	Form ES-401-1

SYSTEM #/NAME	K1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	TOPIC(S)	IMP	#
218000 ADS CFR		01										Describe the relationship between ADS Logic power and the operation of the ADS logic.	3.3	39 872
223002 PCIS / Nuclear Steam Supply Shutoff CFR								03				Determine the operator actions required to mitigate a NSSSS logic failure.	3.3	40 873
239002 SRVs CFR				09								*Describe the design features available to determine if a SRV is open.*	3.6	41 874
259002 Reactor Water Level Control CFR								04				Describe the operator response to a failure of RFPT speed control with speed rising.	3.1	42 875
259002 Reactor Water Level Control CFR										06		Describe prerequisites for transferring the Feedwater system to 3-element control.	3.2	43 233a
261000 SGTS CFR									03			Describe the SGTS damper logic following system initiation.	2.9	44 876
262001 AC Electrical Distribution CFR						01						Given plant conditions and a partial loss of DC power, determine the affect to the AC distribution system.	3.4	45 877
262002 UPS (AC/DC) CFR				01								Given plant conditions and degraded AC power, determine the status of plant inverters.		46 878
263000 DC Electrical Distribution CFR				01								Given a loss of AC power to battery chargers, determine the affects to the DC distribution system.	3.4	47 879
264000 EDGs CFR								10				Describe EDG response to a LOCA.	4.2	48 880
264000 EDGs CFR											2.4.48	Determine EDG status from control room alarms and indications and any required operator actions to improve plant conditions.	3.8	49 881
300000 Instrument Air CFR			01									Determine the effect on the plant given a loss of Instrument Air to the containment.	2.9	50 882
PAGE 2 TOTAL TIER 2 GROUP 1	0	1	1	3	0	1	0	3	1	1	1	PAGE TOTAL # QUESTIONS	12	

PLANT SYSTEMS - TIER 2 GROUP 1 (RO/SRO)

SYSTEM #/NAME	K1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	TOPIC(S)	IMP	#
300000 Instrument Air CFR						13						Determine the affect of a clogged filter on the Instrument Air system.	2.3	51 883
400000 Component Cooling Water CFR	04											Determine the method used to confirm a reactor coolant leak into the CCW system.	3.1	52 884
400000 Component Cooling Water CFR							02					Determine the affect to the plant if the CCW temperature control fails.	2.8	53 885
203000 RHR/LPCI: Injection Mode CFR											2.3.11	Given LOCA conditions, determine how LPCI works in conjunction with the other ECCS to control radiation releases.	3.2 *	86 918
209001 LPCS CFR											2.1.15	Given a short-term problem associated with LPCS that does not affect operability, determine the most effective method to provide the information to operations personnel.	3.0 *	87 919
215003 IRM CFR											2.4.16	Given plant conditions requiring entry into the EOPs and the need to insert the IRMs, determine the correct procedure hierarchy to accomplish the task.	4.0 *	88 920
215004 Source Range Monitor CFR											2.2.21	Given the applicable Tech Specs and a repaired SRM detector, determine the surveillance requirements to ensure operability.	3.5 *	89 921
217000 RCIC CFR											*2.1.10*		*3.9 **	90 922
												* SRO Only Questions		
PAGE 3 TOTAL TIER 2 GROUP 1	1	0	0	0	0	1	1	0	0	0	5	PAGE TOTAL # QUESTIONS	8	
PAGE 1 TOTAL TIER 2 GROUP 1	0	1	0	1	2	0	1	2	0	2	2	PAGE TOTAL # QUESTIONS	11	
PAGE 2 TOTAL TIER 2 GROUP 1	0	1	1	3	0	1	0	3	1	1	1	PAGE TOTAL # QUESTIONS	12	
TIER 2GROUP 1 TOTALS	1	2	1	4	2	2	2	5	1	3	8		31	

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE PLANT SYSTEMS - TIER 2 GROUP 2 (RO/SRO)	Form ES-401-1

SYSTEM #/NAME	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
201001 CRD Hydraulic CFR														
201002 RMCS												N/A GGNS		
201003 Control Rod and Drive Mechanism CFR														
201004 RSCS												N/A GGNS		
201005 RCIS CFR					10							Describe the purpose for the rod withdrawal limiter.	3.3	54 886
201006 RWM												N/A GGNS		
202001 Recirculation CFR											*2.4.11*		*3.6*	55 887
202002 Recirculation Flow Control CFR41.6	01											Given plant conditions, determine any automatic actions associated with the Recirculation System HPUs.	3.6	56 888
204000 RWCU CFR				06								Determine the correct flow path to use RWCU as an alternate shutdown cooling.	2.8	57 889
214000 RPIS												N/A GGNS		
215001 Traversing In-Core Probe CFR														
215002 RBM												N/A GGNS		
PAGE 1 TOTAL TIER 2 GROUP 2	1	0	0	1	1	0	0	0	0	0	1	PAGE TOTAL # QUESTIONS	4	

PLANT SYSTEMS - TIER 2 GROUP 2 (RO/SRO)

SYSTEM #/NAME	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
216000 Nuclear Boiler Instrumentation CFR														
219000 RHR /LPCI Suppression Pool Cooling Mode CFR														
223001 Primary CTMT and Auxiliaries CFR	08											Determine the limitations to SRV usage given a reduced suppression pool level.	3.8	58 890
226001 RHR/LPCI: CTMT Spray Mode CFR														
230000 RHR/LPCI: Torus/Pool Spray Mode												N/A GGNS		
233000 Fuel Pool Cooling and Cleanup CFR														
234000 Fuel Handling Equipment CFR														
239001 Main and Reheat Steam CFR			04									Given plant conditions including a MSIV closure, determine the affect to the Offgas system.	2.8	59 891
239003 MSIV Leakage Control CFR	02											Explain the relationship between the MSIV Leakage Control system and SGTS.	3.0	60 892
241000 Reactor/Turbine Pressure Regulator CFR											2.4.6	Describe the bases for each of the Scram ONEP immediate actions.	4.0	61 893
PAGE 2 TOTAL TIER 2 GROUP 2	2	0	1	0	0	0	0	0	0	0	1	PAGE TOTAL # QUESTIONS	4	

SYSTEM #/NAME	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	TOPIC(S)	IMP	#
245000 Main Turbine Gen./Aux. CFR									02			Determine main turbine critical speeds as it is rolled to rated speed.	2.8	62 894
256000 Reactor Condensate CFR														
259001 Reactor Feedwater CFR								03				Determine necessary actions and priorities immediately after a single condensate pump trips with the plant at rated conditions.	3.6	63 895
268000 Radwaste CFR	04											Determine the Drywell Floor Drains indications available to detect drywell general area leakage.	2.9	64 896
271000 Offgas CFR														
272000 Radiation Monitoring CFR														
286000 Fire Protection CFR														
288000 Plant Ventilation CFR														
290001 Secondary CTMT CFR				03								Determine inputs to the Fuel Pool leak detection standpipe.	2.8	65 897
290003 Control Room HVAC CFR														
290002 Reactor Vessel Internals CFR											2.4.14	Given a severe accident condition, describe the bases for why the transition is made from the EOPs to the SAPs.	3.9 *	91 923
226001 RHR/LPCI: CTMT Spray Mode CFR								13				Determine the affects to the Containment Spray mode of RHR given a valve interlock failure.	2.9 *	92 924
234000 Fuel Handling Equipment CFR								01				Determine the affects to fuel handling operations given a Refueling Bridge interlock failure.	3.7 *	93 925
												* SRO Only Questions		
PAGE 3 TOTAL TIER 2 GROUP 2	1	0	0	1	0	0	0	3	1	0	1	PAGE TOTAL # QUESTIONS	7	
PAGE 1 TOTAL TIER 2 GROUP 2	1	0	0	1	1	0	0	0	0	0	1	PAGE TOTAL # QUESTIONS	4	
PAGE 2 TOTAL TIER 2 GROUP 2	2	0	1	0	0	0	0	0	0	0	1	PAGE TOTAL # QUESTIONS	4	
TIER 2 GROUP 2														

TOTALS	4	0	1	2	1	0	0	3	1	0	2		15	

Facility: **Grand Gulf Nuclear Station** Date of Exam: **12 August 2005**

Category	**K/ A#**	**Topic**	**SRO**		**SRO-Only**	
			IR	**#**	**IR**	**#**
1. Conduct Of Operations	2.1.19	Given plant conditions and the PDS computer, determine necessary actions based on PBDS counts.	3.0	66 898		
	2.1.25	Given plant conditions and EP3 graphs, determine the correct mitigation strategy.	3.1	67 899		
	2.1.29	Determine the correct locking device color coding for locked components.	3.3	68 237a		
	2.1.2	Given conditions, determine when an act of sabotage or tampering should be suspected.			4.0	94 926
	2.1.12				*4.0*	95 927
	Subtotal			3		2
2. Equipment Control	2.2.1	Given plant conditions, determine proper operation of the IRMs.	3.6	69 900		
	2.2.30	Discuss the duties of the operator assigned to communicate with the refueling floor SRO during core alterations.	3.3	70 901		
	2.2.19	Describe the process for generating a maintenance work request.			3.1	96 928
	2.2.16	Determine who is responsible for reviewing the installation and removal of temporary alterations.			2.6	97 929
	Subtotal			2		2
3. Radiation Control	2.3.1	Given the need to enter a high radiation area, determine the allowed time in the area to prevent exceeding the administrative exposure limits.	3.0	71 902		
	2.3.4	Given plant conditions and applicable Emergency Planning Procedures, determine the radiation exposure limits that are in effect.	3.1	72 903		
	2.3.6	Given liquid radwaste batch release data, determine which does not require Operations approval or a discharge permit.			3.1	98 930
	Subtotal			2		1
4. Emergency Procedures / Plan	2.4.20	Given plant conditions, determine the bases for any applicable EOP cautions.	4.0	73 904		
	2.4.25	Given plant conditions including a fire, determine the proper response.	3.4	74 905		
	2.4.43	Given plant conditions and Emergency Planning Procedures, determine the available emergency communications systems.	3.5	75 906		
	2.4.47	Given plant conditions and indications from the recirculation pump shaft seals, analyze the condition and determine the probable failure mechanism.			3.7	99 931
	2.4.44	Given plant conditions that warrant a General Emergency, determine the correct protective action recommendations.			4.0	100 932
	Subtotal			3		2
Tier 3 Point Total				10		7

Tier/ Group	Randomly Selected K/A	Reason for Rejection
2/1	206000	High Pressure Core Injection (HPCI) – GGNS does not have a HPCI System for water inventory control.
2/1	207000	Isolation (Emergency) Condenser – GGNS does not have an Isolation Condenser for pressure suppression.
2/2	201002	Reactor Manual Control System (RMCS) – GGNS utilizes the BWR 6 Rod Control and Information System.
2/2	201004	Reactor Sequence Control System (RSCS) – GGNS utilizes the BWR 6 Rod Control and Information System.
2/2	201006	Rod Worth Minimizer (RWM) – GGNS utilizes the BWR 6 Rod Control and Information System.
2/2	214000	Rod Position Information System (RPIS) – GGNS utilizes the BWR 6 Rod Control and Information System.
2/2	215002	Rod Block Monitor (RBM) – GGNS utilizes the BWR 6 Rod Control and Information System.
2/2	230000	RHR/LPCI: Torus/Pool Spray Mode – GGNS does not have a Torus/Pool Spray mode of the RHR System.
1/1	600000	Plant Fire On Site – AK3 was selected for the Random Selection topic. This topic has 4 statements of which only AK3.04 has an importance of > 2.5 for a RO.
2/1	203000	RHR/LPCI: Injection Mode – K5.01 was eliminated due to the testable check valves for RHR are being disabled via plant change per an ER, thus the selection resulted in the only other K/A K5.02.
2/1	300000	Instrument Air – K6.04 has an importance of > 2.5, however GGNS does not have a Service Air Refusal Valve.
2/2	215001	Traversing In-Core Probe – At GGNS TIPS is only operated by Reactor Engineers. The only operations involvement with the system is protective tagging.
1/1	295027 295038	High Containment Temperature High Offsite Release Rate – Random selection for Generics was 2.2.34 Knowledge of the process for determining the internal and external effects on core reactivity. This K/A does not apply to these two Emergency/Abnormal Plant Evolutions. Random selection was redrawn.
3	Generics	Random selection of 2.2.31 Knowledge of procedures and limitations involved in initial core loading is not applicable to GGNS.

1/1	295027	High Containment Temperature – Generics was selected for the topical area. 2.2.18 Knowledge of the process for managing maintenance activities during shutdown operations and 2.2.1 Ability to perform pre-startup procedures for the facility/including operating those controls associated with plant equipment that could affect reactivity, do not apply to high containment temperature in the realm of emergency level. There is not sufficient energy to raise Containment Temperature to the emergency level.
1/1	295001	Partial or Complete Loss of Forced Core Flow Circulation The initial random selection selected 2.3.2 Alara considerations. Section 2.3 of the Generics have limited applicability for this evolution. Attachment 2 section 1 sentence 4 allows elimination of these K/As without justification. The K/As listed in sentence 1 were numbered 1 – 16 and randomly selected to apply K/A 2.4.4.
1/1	*600000*	*Initial topic called for AA2.16 Ability to determine and interpret as applied to a plant fire on site – vital equipment and control systems to be maintained and operated during a fire. The original outline identified the Main Transformers. The Main Transformers are only vital to Main Generator output. K/A mismatch was felt to be the case and changed the topic to basis for separating vital equipment from the Main Control Room during a fire in the Main Control Room to better fit the K/A based on 10CFR 50 Appendix R considerations.*
1/1	*295025*	*Initial K/A was K3.05, this is similar to K/A for 217000 A1.02. Reselected to 295025 K3.06*
2/2	*202001*	*Initial K/A was Generic 2.2.25 Tech Spec Bases. Not considered RO level knowledge. Reselected to Generic 2.4.11.*
3 SRO	*2.1.24*	*Protective Tagging is considered an RO function for written examinations. Reselected to 2.1.12 for SRO Only question.*
2/1 SRO	*217000*	*Initial K/A was 2.1.25 which is RO level knowledge. Reselected to 2.1.10 Knowledge of facility License and conditions which is SRO level knowledge.*

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **1** Op-Test No.: **Day 1**

Examiners: ____________________ Operators: ____________________

____________________ ____________________

____________________ ____________________

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

1. Start RCIC for testing per *EPI* CST to CST.
2. Respond to a failure of 1C34-LI-R606C RPV Narrow Range Level 'C' downscale.
3. Take actions in response to a Low Pressure Feedwater Heater 3C Tube leak and Failure of the Heater String to Isolate. Complete actions of the Loss of Feedwater Heating ONEP and Reduction in Recirculation System Flowrate ONEP.
4. Respond to a trip of RCIC.
5. Respond to a loss of RPS normal power supply.
6. Take actions for a double Recirculation Pump downshift to manually scram the reactor.
7. Take actions per the EOPs in response to an ATWS and mitigate the consequences of the ATWS with Main Steam Bypass Valves.
8. Respond to a failure of Division II ECCS to manually initiate via the Manual Initiation pushbutton.

Initial Conditions: Reactor Power is at 100 %.

INOPERABLE Equipment

SRMs 'E' & 'F' are INOP

APRM 'H' is INOP due to a failed *FCTR* card.

LPCS Pump is tagged out of service for motor oil replacement.

ESF Transformer 12 is tagged out of service Entergy – Mississippi maintenance.

Appropriate clearances and LCOs are written.

Turnover: The plant is operating at 100% power. Operate RCIC CST to CST at rated flow per a controlled startup in the *EPI* to allow taking of engineering data with RCIC operating 800 gpm at 1000 psig Standby Service Water 'A' is operating. *Containment Ventilation is operating in High Volume Purge.* There are scattered thundershowers reported in the Tensas Parish area.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP)	Start RCIC and operate CST to CST per *EPI.* *(EPI 04-1-03-E51-2)*

Scenario **1** Day **1** (Continued)

Event No.	Malf. No.	Event Type*	Event Description
2	**1** fw126c@ 0	TS (SS)	Respond to RPV Narrow Range Level 'C' instrument failure downscale. Complete **Technical Specification** determination.
3	**2** fw232i @ 50% ramp to 80%	R (RO)	Respond to a tube failure in LP FW Heater 3C. Perform actions per ONEP 05-1-02-V-5 and ONEP 05-1-02-III-3. Lower Reactor power with Recirc flow.
		C (BOP)	With a failure to isolate the Condensate System. Perform actions per ARI 04-1-02-1H13-P870 6A-B3 to isolate LP Feedwater Heater String 'C'.
4	**3** e51047	C (BOP) TS (SS)	RCIC Turbine Trip. Complete **Technical Specification** determination.
5	**4** c71077b	C (RO/ BOP)	Respond to a RPS 'B' Motor Generator EPA Breaker Trip per the ONEP 05-1-02-III-2.
6	**5** fw201; c71076	C (RO)	Respond to a double Reactor Recirculation Pump down shift, Automatic RPS actuation fails requiring insertion of a manual Reactor Scram.
7	**6** c11164 @ 0.2%	M (ALL)	Upon Reactor Scram recognize the failure of all control rods to fully insert and take actions per EOPs for ATWS with Main Steam Bypass Valves.
	7 di_1e12 m617@ NORM	I (BOP)	Upon orders to initiate and override Low Pressure ECCS, recognize the failure of Division II to initiate via Manual Initiation pushbutton. Take actions upon automatic initiation to override Division II Low Pressure ECCS.
* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor			

Critical Tasks

- Terminate and prevent injection from Feedwater and ECCS as required.
- Commence injection into the reactor using Feedwater or RHR ‘A’ or ‘B’ through Shutdown Cooling to restore and maintain level > -192 inches.
- Insert Control Rods in response to ATWS conditions.

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **2** Op-Test No.: **Day 1**

Examiners: ____________________ Operators: ____________________

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

1. Start SSW 'A' in support of chemical addition.
2. Raise Reactor Power by withdrawing control rods. Respond to single control rod drift per ONEP 05-1-02-IV-1.
3. Respond to ESF Transformer 21 trouble and subsequent trip with a failure of DG 12 to start.
4. *Respond to Main Generator TVR failure.*
5. Take actions to mitigate a large break failure of Feedwater piping in the Drywell per EOPs. (LOCA is NOT severe enough to result in depressurization of RPV.)
6. Respond to a failure of Division 1 ECCS to automatically initiate on High Drywell Pressure.
7. Respond to a failure of High Pressure Core Spray to inject. (LOCA with degraded high pressure sources.)

Initial Conditions: Reactor Power is at 45 %. Plant startup is in progress following an outage. Reactor Recirculation pumps in Fast Speed; a single Reactor Feed Pump in Three element Master Level Control; both Heater Drain Pumps are pumping forward.

INOPERABLE Equipment

SRMs 'E' & 'F' are INOP and bypassed.
APRM 'H' is INOP due to a failed *FCTR* card.
LPCS Pump is tagged out of service for pump seal replacement.
ESF 12 Transformer is tagged out of service for maintenance.
Appropriate clearances and LCOs are written.

Turnover: Chemistry requires SSW 'A' in operation to support a chemical addition. Continue plant startup per IOI-2. There are scattered thunder showers reported in the Tensas Parish area.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP)	Place Standby Service Water 'A' in service for chemical addition. (EPI 04-1-03-P41-1)
2		R (RO)	Raise Reactor power using control rods to 49%. (Control Rod Pull Sheet)
3	**1** z161161_24_17	C (RO) TS (SS)	Respond to single control rod drift taking actions to insert the control rod. (ONEP 05-1-02-IV-1) Disarm Control Rod. Complete **Technical Specification** determination.

Scenario **2** Day **1** (Continued)

Event No.	Malf. No.	Event Type*	Event Description
4	**2** p807_4a_f_2 ON r21180 n41140b	C (BOP) TS (SS)	Respond to trouble and trip of ESF Transformer 21 with a failure of DG 12 to Start. Complete **Technical Specification** determination. (ONEP 05-1-01-I-4)
5	***3*** *n41102*	*C (RO)*	*Respond to a failure of the Main Generator Voltage Regulator. (ARI 04-1-02-1H13-P680 9A-C15 and SOI 04-1-01-N40-1)*
6	**4** fw0171b @ 70% rr063b @ 1% ramp to 4%	M (ALL)	Respond to indications of large break LOCA on Feedwater Line 'B' per EOPs. (B21-F065B will close if attempted.)
	5 rr040e@ 0 rr041e @ 83%	I (BOP)	Respond to a failure of Division 1 ECCS to automatically initiate on High Drywell Pressure.
	6 e22159a@0	C (BOP)	Respond to a failure of High Pressure Core Spray to inject.

* (N)ormal, (R)eactivity, (I)nstrument, (C)omponent, (M)ajor

Critical Tasks

- Recognize failure of Division 1 to initiate and manually initiate Division 1.
- Isolate the failed Feedwater line and re-establish Condensate/Feedwater or when RPV level reaches -160 inches wide range, Emergency Depressurizes the RPV to allow injection from Low Pressure systems (if level cannot be restored and maintained above -192 inches).