Facility: **Grand Gulf Nuclear Station** Date of Examination: **16 August 2005**
Examination Level (circle one) **RO** / SRO Operating Test Number:

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	D; S	Given plant conditions, perform the Idle Loop Startup Surveillance for Recirculation System. GJPM-RO-ADM-1 K/A 2.1.20: 4.3 Safety Function 1
Conduct of Operations	M; S	Given plant conditions, complete documentation for Shift Turnover GJPM-RO-ADM-2 K/A 2.1.3: 3.0
Equipment Control	M	Given a component to be isolated for a work order, prepare a tagout using the eSOMS program. GJPM-RO-ADM-3 K/A 2.2.13: 3.6
Radiation Control	D	Complete entry and egress from the CAA with access requirements for a High Radiation Area. GJPM-OPS-ADM-26 K/A 2.3.1: 2.6; 2.3.4: 2.5; 2.3.5: 2.3
Emergency Plan	**N/A**	**N/A**

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

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

Facility: **Grand Gulf Nuclear Station** Date of Examination: **15 August 2005**
Exam Level (circle one) **RO** / SRO-I / SRO-U Operating Test Number:

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

System / JPM Title	Type Code*	Safety Function
a. 202001 Recirculation System - Startup idle Recirculation Pump <30% power with incomplete start actuation.	S; M; A	4
b. 201001 Control Rod Drive Hydraulic System - Rotate operating CRD pumps, trip of newly operating pump.	S; M; A	1
c. 259001 Reactor Feed Water System - Startup Second Reactor Feed Pump and place on Master level control, with failure of Automatic controller.	S; N; A	2
d. 226001 RHR Containment Spray - Secure Containment Sprays and align for injection to RPV with failure of one RHR injection valve.	S; N; A	5 ESF
e. 264000 Emergency Diesel Generators – Start, parallel, and load the Diesel Generator with trip of SSW.	S; D; A	6 ESF
f. 290003 Control Room HVAC System – Secure Control Room Standby Fresh Air System.	C; D	9 ESF
g. 239001 Main & Reheat Steam System - Open Main Steam Isolation Valves.	S; D; L	3 ESF
h. 201005 Rod Control & Information System – Operate RCIS to bring the reactor critical.	S; D; L	7
In-Plant Systems@ (3 for RO; 3 for SRO-I; 3or2 for SRO-U)		
i. 295003 Partial Loss of AC – Reset undervoltage lockouts on buses when power is restored, with one lockout failing to reset.	R; M; E; A	6
j. 286000 Fire Protection System – Manually initiate fire suppression for the Standby Gas Filter Train with the train operating.	R; N; E	8
k. 295016 Control Room Abandonment – Startup RHR in Suppression Pool Cooling.	N; E; L	7 ESF

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

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

Facility: **Grand Gulf Nuclear Station** Date of Examination: **16 August 2005**
Examination Level (circle one) RO / **SRO** Operating Test Number:

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M	Given a completed AC/DC Lineup following a failure of a Diesel Generator, determine the LCO action requirements and generate an eSOMS LCO. GJPM-SRO-ADM-1 K/A 2.1.12: 4.0 Safety Function 6
Conduct of Operations	M	Given a failed relay, determine the impact on plant operations using facility drawings. GJPM-SRO-ADM-2 K/A 2.1.24: 3.1
Equipment Control	M	Given a work order and prepared tagout, determine the adequacy of the tagout and the impact on plant operations. GJPM-SRO-ADM-3 K/A 2.2.13: 3.8; 2.2.17: 3.5
Radiation Control	N	Given plant conditions, determine Protective Action Recommendations and Radiological Considerations for On-Site Personnel. GJPM-SRO-A&E-41 K/A 2.3.8: 3.2
Emergency Plan	M	Given plant conditions, determine entry into the Site Emergency Plan and complete the initial notification forms. GJPM-SRO-A&E-42 K/A 2.4.41: 4.1; 2.4.38: 4.0; 2.4.40: 4.0

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

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

Facility: **Grand Gulf Nuclear Station** Date of Examination: **15 August 2005**
Exam Level (circle one) RO / **SRO-I** / SRO-U Operating Test Number:

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

System / JPM Title	Type Code*	Safety Function
a. 202001 Recirculation System - Startup idle Recirculation Pump <30% power with incomplete start actuation.	S; M; A	4
b. 201001 Control Rod Drive Hydraulic System - Rotate operating CRD pumps, trip of newly operating pump.	S; M; A	1
c. 259001 Reactor Feed Water System - Startup Second Reactor Feed Pump and place on Master level control, with failure of Automatic controller.	S; N; A	2
d. 226001 RHR Containment Spray - Secure Containment Sprays and align for injection to RPV with failure of one RHR injection valve.	S; N; A	5 ESF
e. 264000 Emergency Diesel Generators – Start, parallel, and load the Diesel Generator with trip of SSW.	S; D; A	6 ESF
f. 290003 Control Room HVAC System – Secure Control Room Standby Fresh Air System.	C; D	9 ESF
g. 239001 Main & Reheat Steam System - Open Main Steam Isolation Valves.	S; D; L	3 ESF
h. N/A		
In-Plant Systems@ (3 for RO; 3 for SRO-I; 3or2 for SRO-U)		
i. 295003 Partial Loss of AC – Reset undervoltage lockouts on buses when power is restored, with one lockout failing to reset.	R; M; E; A	6
j. 286000 Fire Protection System – Manually initiate fire suppression for the Standby Gas Filter Train with the train operating.	R; N; E	8
k. 295016 Control Room Abandonment – Startup RHR in Suppression Pool Cooling.	N; E; L	7 ESF

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

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

Facility: **Grand Gulf Nuclear Station** Date of Examination: **15 August 2005** Exam Level (circle one) RO / SRO-I / **SRO-U** Operating Test Number:		
Control Room Systems@ (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U, including 1 ESF)		
System / JPM Title	**Type Code***	**Safety Function**
a. 202001 Recirculation System - Startup idle Recirculation Pump <30% power with incomplete start actuation.	S; M; A	4
b. 201001 Control Rod Drive Hydraulic System - Rotate operating CRD pumps, trip of newly operating pump.	S; M; A	1
c. N/A		
d. N/A		
e. N/A		
f. 290003 Control Room HVAC System – Secure Control Room Standby Fresh Air System.	C; D	9 ESF
g. N/A		
h. N/A		
In-Plant Systems@ (3 for RO; 3 for SRO-I; 3or2 for SRO-U)		
i. N/A		
j. 286000 Fire Protection System – Manually initiate fire suppression for the Standby Gas Filter Train with the train operating.	R; N; E	8
k. 295016 Control Room Abandonment – Startup RHR in Suppression Pool Cooling.	N; E; L	7 ESF

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

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

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE EMERGENCY & ABNORMAL PLANT EVOLUTIONS - TIER 1 GROUP 1 (RO/SRO)							Form ES-4
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
295003 Partial or Complete Loss of AC Power/ 6 CFR			01				Given plant conditions and a loss of AC power, determine the necessary actions to restore vital busses.	3.3
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 SSW system.	3.8
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
295006 SCRAM / 1 CFR	02						Given plant conditions following a reactor scram, determine if adequate shutdown margin exists.	3.4
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
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
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
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
295023 Refueling Accidents / 8 CFR	03						Determine the correct operator response to inadvertent criticality following a refueling accident.	3.7
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
295025 High Reactor Pressure / 3 CFR			05				Describe RCIC operation following a reactor scram where the SRVs are used to control reactor pressure.	3.6
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
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
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-4	
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
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
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
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
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
600000 Plant Fire On Site / 8			04				Determine the required procedural actions for a fire on the plant site.	2.8
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)**							Form ES-4
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
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
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
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-4	
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	
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	
295035 Secondary Containment High Differential Pressure / 5 CFR	02						Given accident conditions and a Standby Gas Treatment system failure, determine the type of release.		
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	
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	

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE PLANT SYSTEMS - TIER 2 GROUP 1 (RO/SRO)												Form ES-40	
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	8
205000 Shutdown Cooling CFR				04								Describe the RHR Shutdown Cooling system NPSH interlocks.	2.6	8
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	8
209002 HPCS CFR										09		Describe available methods to raise/lower suppression pool level using HPCS.	3.4	8
209002 HPCS CFR											2.1.28	Describe the bases for the HPCS injection valve high reactor water level interlock.	3.2	8
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	8
212000 RPS CFR								12				Given plant conditions including a partial main turbine stop/control valve closure, determine the effect to RPS.	4.0	8
215003 IRM CFR					03							Describe the reason for the precaution concerning driving IRMs during surveillance activities.	3.0	8
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	8
215005 APRM / LPRM CFR		02										Given a partial loss of plant electrical power, determine the effect to the APRMs.	2.6	8
217000 RCIC CFR							02					Predict how a reactor pressure change will affect RCIC system flow.	3.3	8
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-40	
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	8
223002 PCIS / Nuclear Steam Supply Shutoff CFR								03				Determine the operator actions required to mitigate a NSSSS logic failure.	3.0	8
239002 SRVs CFR				09								Describe the design features available to determine if a SRV is open.	3.7	8
259002 Reactor Water Level Control CFR								04				Describe potential problems associated with operating a RFP in Emergency Manual	3.0	8
259002 Reactor Water Level Control CFR										06		Describe prerequisites for transferring the Feedwater system to 3-element control.	3.1	2
261000 SGTS CFR									03			Describe the SGTS damper logic following system initiation.	3.0	8
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	8
262002 UPS (AC/DC) CFR				01								Given plant conditions and degraded AC power, determine the status of plant inverters.	3.1	8
263000 DC Electrical Distribution CFR				01								Given a loss of AC power to battery chargers, determine the affects to the DC distribution system.		8
264000 EDGs CFR								10				Describe EDG response to a LOCA.	3.9	8
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	8
300000 Instrument Air CFR			01									Determine the effect on the plant given a loss of Instrument Air to the containment.	2.7	8
PAGE 2 TOTAL TIER 2 GROUP 1	0	1	1	3	0	1	0	3	1	1	1	PAGE TOTAL # QUESTIONS	12	

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

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
400000 Component Cooling Water CFR	04											Determine the method used to confirm a reactor coolant leak into the CCW system.	2.9
400000 Component Cooling Water CFR							02					Determine the affect to the plant if the CCW temperature control fails.	2.8
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

Form ES-4

GRAND GULF NUCLEAR STATION

BWR EXAMINATION OUTLINE

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

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
201006 RWM												N/A GGNS	
202001 Recirculation CFR											2.2.25	Given degraded plant conditions determine any applicable Recirculation Loop LCOs.	2.5
202002 Recirculation Flow Control CFR41.6	01											Given plant conditions, determine any automatic actions associated with the Recirculation System HPUs.	3.5
204000 RWCU CFR				06								Determine the correct flow path to use RWCU as an alternate shutdown cooling.	2.6
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-
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
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
239003 MSIV Leakage Control CFR	02											Explain the relationship between the MSIV Leakage Control system and SGTS.	2.9
241000 Reactor/Turbine Pressure Regulator CFR											2.4.6	Describe the bases for each of the Scram ONEP immediate actions.	3.1
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-
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
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
268000 Radwaste CFR	04											Determine the Drywell Floor Drains indications available to detect drywell general area leakage.	2.7
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
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**		**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 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-4
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
295003 Partial or Complete Loss of AC Power/ 6 CFR			01				Given plant conditions and a loss of AC power, determine the necessary actions to restore vital busses.	3.5
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 SSW system.	4.1
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
295006 SCRAM / 1 CFR	02						Given plant conditions following a reactor scram, determine if adequate shutdown margin exists.	3.7
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
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
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
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
295023 Refueling Accidents / 8 CFR	03						Determine the correct operator response to inadvertent criticality following a refueling accident.	4.0
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
295025 High Reactor Pressure / 3 CFR			05				Describe RCIC operation following a reactor scram where the SRVs are used to control reactor pressure.	3.7
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
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
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-4
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
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
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
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
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
600000 Plant Fire On Site / 8			04				Determine the required procedural actions for a fire on the plant site.	3.4
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*
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*
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*
295027 High Containment Temperature / 5 CFR						2.2.22	Explain the bases for the Technical Specification Containment average air temperature limit.	4.1*
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*
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*
600000 Plant Fire On Site / 8					16		Explain the automatic response of the plant Fire Protection system to a main transformer fire.	3.5*
							* 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

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE EMERGENCY & ABNORMAL PLANT EVOLUTIONS - TIER 1 GROUP 2 (RO/SRO)						Form ES-4	
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
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
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
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-4
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
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
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
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
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*
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*
295020 Inadvertent Cont. Isolation / 5 & 7 CFR					03		Given a partial MSIV closure, determine the affect on reactor power.	3.7*
							* 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

GRAND GULF NUCLEAR STATION	BWR EXAMINATION OUTLINE PLANT SYSTEMS - TIER 2 GROUP 1 (RO/SRO)											Form ES-401-	
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
205000 Shutdown Cooling CFR				04								Describe the RHR Shutdown Cooling system NPSH interlocks.	2.6
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
209002 HPCS CFR										09		Describe available methods to raise/lower suppression pool level using HPCS.	3.5
209002 HPCS CFR											2.1.28	Describe the bases for the HPCS injection valve high reactor water level interlock.	3.3
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
212000 RPS CFR								12				Given plant conditions including a partial main turbine stop/control valve closure, determine the effect to RPS.	4.1
215003 IRM CFR					03							Describe the reason for the precaution concerning driving IRMs during surveillance activities.	3.1
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
215005 APRM / LPRM CFR		02										Given a partial loss of plant electrical power, determine the effect to the APRMs.	2.8
217000 RCIC CFR							02					Predict how a reactor pressure change will affect RCIC system flow.	3.3
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-40
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
223002 PCIS / Nuclear Steam Supply Shutoff CFR								03				Determine the operator actions required to mitigate a NSSSS logic failure.	3.3
239002 SRVs CFR				09								Describe the design features available to determine if a SRV is open.	3.6
259002 Reactor Water Level Control CFR								04				Describe potential problems associated with operating a RFP in Emergency Manual	3.1
259002 Reactor Water Level Control CFR										06		Describe prerequisites for transferring the Feedwater system to 3-element control.	3.2
261000 SGTS CFR									03			Describe the SGTS damper logic following system initiation.	2.9
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
262002 UPS (AC/DC) CFR				01								Given plant conditions and degraded AC power, determine the status of plant inverters.	
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
264000 EDGs CFR								10				Describe EDG response to a LOCA.	4.2
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
300000 Instrument Air CFR			01									Determine the effect on the plant given a loss of Instrument Air to the containment.	2.9
PAGE 2 TOTAL TIER 2 GROUP 1	0	1	1	3	0	1	0	3	1	1	1	PAGE TOTAL # QUESTIONS	12

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

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
400000 Component Cooling Water CFR	04											Determine the method used to confirm a reactor coolant leak into the CCW system.	3.1
400000 Component Cooling Water CFR							02					Determine the affect to the plant if the CCW temperature control fails.	2.8
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 *
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 *
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 *
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 *
217000 RCIC CFR											2.1.25	Given plant conditions and procedures, determine Suppression Pool Level using the RCIC System.	3.1 *
												* 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-

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	
201006 RWM												N/A GGNS		
202001 Recirculation CFR											2.2.25	Given degraded plant conditions determine any applicable Recirculation Loop LCOs or safety limits.	3.7	
202002 Recirculation Flow Control CFR41.6	01											Given plant conditions, determine any automatic actions associated with the Recirculation System HPUs.	3.6	
204000 RWCU CFR				06								Determine the correct flow path to use RWCU as an alternate shutdown cooling.	2.8	
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-40

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
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
239003 MSIV Leakage Control CFR	02											Explain the relationship between the MSIV Leakage Control system and SGTS.	3.0
241000 Reactor/Turbine Pressure Regulator CFR											2.4.6	Describe the bases for each of the Scram ONEP immediate actions.	4.0
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-
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)	IMI
245000 Main Turbine Gen./Aux. CFR									02			Determine main turbine critical speeds as it is rolled to rated speed.	2.8
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
268000 Radwaste CFR	04											Determine the Drywell Floor Drains indications available to detect drywell general area leakage.	2.9
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
290003 Control Room HVAC CFR													
290002 Reactor Vessel Internals CFR											2. 4. 1 4	Given a severe accident condition, describe the bases for why the transition is made from the EOPs to the SAPs.	3.9 *
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 *
234000 Fuel Handling Equipment CFR								01				Determine the affects to fuel handling operations given a Refueling Bridge interlock failure.	3.7 *
												* 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			
	2.1.25	Given plant conditions and EP3 graphs, determine the correct mitigation strategy.	3.1			
	2.1.29	Determine the correct locking device color coding for locked components.	3.3			
	2.1.2	Given conditions, determine when an act of sabotage or tampering should be suspected.			4.0	
	2.1.24	Given the need to generate a protective tagging clearance, discuss any procedural guidance for use of electrical and mechanical drawings.			3.1	
	Subtotal			3		2
2. Equipment Control	2.2.1	Given plant conditions, determine proper operation of the IRMs.	3.6			
	2.2.30	Discuss the duties of the operator assigned to communicate with the refueling floor SRO during core alterations.	3.3			
	2.2.19	Describe the process for generating a maintenance work request.			3.1	
	2.2.16	Determine who is responsible for reviewing the installation and removal of temporary alterations.			2.6	
	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			
	2.3.4	Given plant conditions and applicable Emergency Planning Procedures, determine the radiation exposure limits that are in effect.	3.1			
	2.3.6	Given liquid radwaste batch release data, determine which does not require Operations approval or a discharge permit.			3.1	
	Subtotal			2		1
4. Emergency Procedures / Plan	2.4.20	Given plant conditions, determine the bases for any applicable EOP cautions.	4.0			
	2.4.25	Given plant conditions including a fire, determine the proper response.	3.4			
	2.4.43	Given plant conditions and Emergency Planning Procedures, determine the available emergency communications systems.	3.5			
	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	
	2.4.44	Given plant conditions that warrant a General Emergency, determine the correct protective action recommendations.			4.0	
	Subtotal			3		2
Tier 3 Point Total				10		7

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:

2 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 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 4. Respond to a trip of RCIC.

5 5. Respond to a loss of RPS normal power supply.

6 6. Take actions for a double Recirculation Pump downshift to manually scram the reactor.

7 7. Take actions per the EOPs in response to an ATWS and mitigate the consequences of the ATWS with Main Steam Bypass Valves.

8 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 power supply 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 when conditions require entry into Level/Power Control.
- 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: ____________________

____________________ ____________________

____________________ ____________________

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

3 1. Start SSW 'A' in support of chemical addition.

4 2. Raise Reactor Power by withdrawing control rods. Respond to single control rod drift per ONEP 05-1-02-IV-1.

5 3. Respond to ESF Transformer 21 trouble and subsequent trip with a failure of DG 12 to start.

6 4. 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.)

7 5. Respond to a failure of Division 1 ECCS to automatically initiate on High Drywell Pressure.

8 6. 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 power supply 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 between 40 and 45%. (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 n41141b	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** 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.)
	4 rr040e@ 0 rr041e @ 83%	I (BOP)	Respond to a failure of Division 1 ECCS to automatically initiate on High Drywell Pressure.
	5 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).

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **3** Op-Test No.: **BACK UP**

Examiners: ______________________ Operators: ______________________

______________________ ______________________

______________________ ______________________

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

2 1. Start 2nd Condensate Pump and Condensate Booster Pump.
3 2. Raise Reactor Power/ Pressure by withdrawing control rods.
4 3. Respond to a stuck control rod.
5 4. Respond to a trip of LCC 15BA3.
6 5. Respond to an automatic and manual scram function failure; ATWS ARI/RPT will insert control rods with *three* control rods stuck withdrawn.
7 6. Recognize the failure of MSIVs to completely isolate and take actions to isolate the Main Steam Lines.
8 7. Recognize and respond to a steam leak in the Auxiliary Building Steam Tunnel. Take actions for mitigation of the leak with a failure of the MSIVs to fully isolate.
9 8. Take actions per the EOPs in response to three stuck control rods following a Reactor Scram.

Initial Conditions: Reactor Power is at 1 % plant heatup and pressurization is in progress. The Reactor is ≈ 400 psig with 1 Condensate and Condensate Booster Pump in service on Startup Level Control. Step 80 of the Control Rod Movement Sequence.

INOPERABLE Equipment

APRM 'H' is INOP due to a failed power supply card
LPCS Pump is tagged out of service for motor oil replacement and will be returned to service in two (2) hours.
ESF-12 Transformer is tagged out of service for Entergy – Mississippi maintenance.

Appropriate clearances and LCOs are written.

Turnover: Continue power ascension. Ready to Start second Condensate and Condensate Booster Pump. There are scattered thundershowers reported in the Tensas Parish area.

Scenario **3 Backup** (Continued)

Event No.	Malf. No.	Event Type*	Event Description
1		N (RO)	Start 2nd Condensate and Condensate Booster Pumps. (SOI 04-1-01-N19-1)
2		R (RO)	Raise reactor power and pressure by withdrawing control rods. (IOI 03-1-01-1 and Control Rod Movement Sheet)
3	**1** z022022 _ 40_45	C (RO/ BOP) TS (SS)	Respond to a stuck control rod during withdrawal. (ONEP 05-1-02-IV-1) Complete **Technical Specification** determination.
4	**2** r21142t	C (BOP/ RO)	Respond to a trip of Load Control Center 15BA3. (ONEP 05-1-02-I-4; 05-1-02-III-5; and SOI 04-1-01-R21-15)
5	**3** c71162	C (RO)	Recognize a failure to scram using RPS and manually scram the reactor using ATWS ARI.
6	**4** ms183b ms184b att9	I (BOP)	Recognize the failure of MSIVs to completely isolate and take actions to isolate the Main Steam Lines. (ONEP 05-1-02-III-5)
7	**5** ms067b @ 20% ms066b @ 0.2% ramp to 10%	M (ALL)	Recognize and respond to a steam leak in the Auxiliary Building Steam Tunnel. Take actions for mitigation of the leak with a failure of the MSIVs to fully isolate.
	6 z022022 _ 36_25 _12_09	C (RO)	Recognize the failure of two additional control rods to fully insert on the Reactor Scram. (Three Rods Out) insert control rods (ONEP 05-1-02-IV-1)

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

Critical Tasks

1 Manually scram the reactor.
2 Isolate the main steam lines.