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	295003 K2.05	GGNS does not have an isolation condenser. Randomly drew K2.04 as replacement.
1/1	295006 K3.05	GGNS does not have a direct turbine/generator trip from a scram. Randomly drew K3.01 as replacement.
1/1	295006 2.1.7	Ability to make accurate/concise verbal reports for a SCRAM will be observed for the operating test. I could not think of a written question that would test this ability. Randomly drew 2.1.10 as replacement
1/1	295038 2.4.30	IR for RO is 2.2, which is less than required 2.5. Then, randomly drew 2.3.5 which has IR 2.3, which is less than required 2.5. Then randomly drew 2.1.12 as replacement.
1/1	295025 A2.05	Could not think of a discriminatory SRO level question for determining/interpreting decay heat generation as it applies to high reactor pressure. Randomly drew A2.03 as replacement.
1/2	295009 A2.03	Could not think of a discriminatory question for determining RWCU blowdown rate as it applies to low water level. Randomly drew A1.03 as replacement.
1/2	295010 K2.04	GGNS does not have a nitrogen make-up system. Randomly drew K2.03 as replacement.
1/2	295011 A2.03	GGNS has no humidity monitoring capability for containment. Randomly drew K2.02 as replacement.
1/2	295022 A1.04	Could not relate loss of CRD pumps to ability to operate/monitor RWCU (plant specific). Randomly drew K3.01 as replacement.
1/2	295032 2.4.44	IR for RO is 2.1, which is less than required 2.5. Then, randomly drew 2.4.38 which has IR 2.2, which is less than required 2.5. Then randomly drew 2.4.5 as replacement.

1/2	295033 2.2.24	Could not relate High Secondary Containment Radiation Levels to use of mechanical and electrical drawings. Randomly drew 2.1.25 as replacement.
2/1	209001 2.2.33	Could not relate 2.2.33, knowledge of control rod programming, to LPCS. Remaining in section 2.2, re-drew 2.2.28, but could not relate knowledge of new and spent fuel movement procedures to LPCS. Redrew 2.2.29, but could not relate SRO fuel handling responsibilities to LPCS. Re-drew 2.2.20, but IR for RO is 2.2, which is less than required 2.5. Then, randomly drew 2.2.15 which has IR for RO of 2.2, which is less than required 2.5. All generics were then made available for selection, and randomly drew 2.1.33 as replacement.
2/1	223002 K2.01	IR for RO is 2.4, which is less than required 2.5. There were no other items under K2. Randomly drew A2.04 as replacement.
2/1	218000 A4	Originally drew A4 category but randomly re-drew A3.02 to meet tier total requirement for A3. A4 had the most questions drawn, so randomly selected 218000 to be changed from A4 to A3.02.
2/1	261000 K1.03	Could not think of a good question to test knowledge of cause/effect relationship between STANDBY GAS TREATMENT SYSTEM and the suppression pool. Randomly re-drew K1.10, but its IR is 2.3 for RO and SRO, which is less than the required 2.5. Randomly re-drew K1.01 as replacement.
2/1	263000 K6.02	This is nearly identical to 263000 A2.02 (effect on DC system by loss of battery room ventilation) drawn for SRO test question #88. Randomly drew K2.01 as replacement.
2/1	264000 K5.03	IR for RO is 2.4, which is less than required 2.5. Randomly drew K5.06 as replacement.
2/1	300000 K1.01	IR for RO is 2.4, which is less than required 2.5. Randomly drew K1.04 as replacement.
2/1	215005 K4.03	IR for RO is 2.1, which is less than required 2.5. Randomly drew K4.02 as replacement.
2/1	215003 2.2.8	IR for RO is 1.8, which is less than required 2.5. Randomly drew 2.1.30 as replacement, but could not think of a discriminating question that would test ability to locate and operate controls including local controls associated with IRMs. Operators only operate IRMs from the reactor control console. Generic 2.1.26 was randomly drawn as replacement, but its IR for RO was 2.2. Randomly drew 2.2.11 as replacement.
2/1	300000 2.3.9	This KA was previously drawn for RO Tier 3 and is too specific to reuse. Randomly drew 2.2.14 as replacement.

2/2	202002 K5.01	Recirculation Flow Control - Fluid coupling is N/A for BWR-6, GGNS. Randomly drew K5.02 as replacement.
2/2	223001 2.2.16	IR for RO is 1.9, which is less than required 2.5. Randomly drew 2.1.4 as replacement, but it was not used. (see reason below)
2/2	223001 2.1.4	Could not think of a question relating shift staffing requirements to primary containment and auxiliaries. Randomly drew 2.4.9 as replacement
2/2	233000 2.1.22	Could not think of a question relating fuel pool cooling and cleanup to determining plant Mode. Randomly drew 2.1.2 as replacement.
2/2	264000 A2.08	Could not think of a discriminatory question at the SRO level for response to initiation of diesel generator fire protection. Randomly drew A2.05 as replacement.
2/2	239001 2.4.20	There are no EOP warnings, cautions, or notes applicable to Main and Reheat Steam. Randomly drew 2.4.6 as replacement.
2/2	241000 K2	All K2 IRs for RO are is less than the required 2.5. Randomly drew K1.14 as replacement.
2/2	268000 K6	First drew K6, all four IRs for RO are less than required 2.5. Then drew K4, but no items are listed under K4. Randomly drew A4.01 as replacement.
2/2	290002 A1	No items listed under A1. Randomly drew generic 2.2.32 as replacement, but its IR for RO is 2.3, which is less than required 2.5. Randomly drew 2.2.22 as replacement.
2/2	202002	This was selected for SRO but had already been selected for tier 2 group 2 for RO portion. Randomly drew 256000 A2.13 as replacement to have broader sample.
2/2	286000	This was selected for SRO but had already been selected for tier 2 group 2 for RO portion. Randomly drew 239001 generic 2.4.20 as replacement to have broader sample.
3	2.1.34	IR for RO is 2.3, which is less than required 2.5. Then randomly drew 2.1.32 as replacement.
3	2.3.3	IR for RO is 1.8, which is less than required 2.5. Then randomly drew 2.3.9 as replacement.
3	2.1.24	Could not write SRO only question for interpreting station electrical and mechanical drawings. Randomly drew 2.1.14 as replacement.

Facility: **Grand Gulf Nuclear Station** Date of Exam: **March 2007**

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	6	2	4				4	3			1	20	5	2	7
	2	0	3	1		N/A		1	1	N/A		1	7	2	1	3
	Tier Totals	6	5	5				5	4			2	27	7	3	10
2. Plant Systems	1	3	3	2	2	2	2	2	2	2	4	2	26	3	2	5
	2	1	0	2	1	1	1	0	1	0	2	3	12	2	1	3
	Tier Totals	4	3	4	3	3	3	2	3	2	6	5	38	5	3	8

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

Note:

1. Ensure that at least two topics from every applicable K/A category are sampled within each tier of the RO and SRO-only outlines (i.e., except for one category in Tier 3 of the SRO-only outline, the "Tier Totals' in each K/A category shall not be less than two).

2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ±1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.

3. Systems/evolutions within each group are identified on the associated outline; systems or evolutions that do not apply at the facility should be deleted and justified; operationally important, site-specific systems that are not included on the outline should be added. Refer to ES-401, Attachment 2 for guidance regarding the elimination of inappropriate K/A statements.

4. Select topics from as many systems and evolutions as possible; sample every system or evolution in the group before selecting a second topic for any system or evolution.

5. Absent a plant-specific priority, only those K/As having an importance rating (IR) of 2.5 or higher shall be selected. Use the RO and SRO ratings for RO and SRO-only portions, respectively.

6. Select SRO topics for Tiers 1 and 2 from the shaded systems and K/A categories.

7.* The generic (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.

8. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IRs) for the applicable license level, and the point totals (#) for each system and category. Enter the group and tier totals for each category in the table above. Use duplicate pages for RO and SRO-only exams.

9. For Tier 3, select topics from Section 2 of the K/A catalog, and enter the K/A numbers, descriptions, IRs, and point totals (#) on Form ES-401-3. Limit SRO selections to K/As that are linked to 10 CFR 55.43.

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	03						Knowledge of operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE CIRCULATION: **thermal limits**	4.1	1
295003 Partial or Complete Loss of AC Power/ 6		04					Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF AC POWER and the following: **AC electrical loads**	3.5	2
295004 Partial or Complete Loss of DC Power / 6		03					Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF DC POWER and the following: **D.C. bus loads**	3.3	3
295005 Main Turbine Generator Trip / 3	03						Knowledge of operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP: **pressure effects on reactor level**	3.7	4
295006 SCRAM / 1			01				Knowledge for the reasons for the following responses as they apply to SCRAM: **reactor water level response**	3.9	5
295016 Control Room Abandonment / 7				07			Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: **Control room/local transfer mechanisms**	4.3	6
295018 Partial or Complete Loss of CCW / 8	01						Knowledge of operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: **effects on component/system operation**	3.6	7
295019 Partial or Complete Loss of Inst. Air / 8			01				Knowledge for the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: **backup air system supply:** plant specific	3.4	8
295021 Loss of Shutdown Cooling / 4			05				Knowledge for the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: **establishing alternate heat removal flow paths**	3.8	9
295023 Refueling Accidents / 8					02		Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS: **fuel pool level**	3.7	10
295024 High Drywell Pressure / 5				02			Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: **HPCS**: plant specific	3.7	11
295025 High Reactor							Ability to operate and/or monitor the		12

Pressure / 3				02			following as they apply to HIGH REACTOR PRESSURE: **reactor/turbine pressure regulating system**	3.8	
295026 Suppression Pool High Water Temp. / 5					02		Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE**: suppression pool level**	3.9	13
295027 High Containment Temperature (Mark III) / 5				02			Ability to operate and/or monitor the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III): **containment ventilation/cooling**	3.5	14
PAGE 1 TOTAL TIER 1 GROUP 1	3	2	3	4	2	0	PAGE TOTAL # QUESTIONS	14	

E/APE #/NAME/SAFETY FUNCTION	K1	K2	K3	A1	A2	G	TOPIC(S)	IMP	#
295028 High Drywell Temperature / 5	01						Knowledge of operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: **reactor water level measurement**	3.7	15
295030 Low Suppression Pool Water Level / 5			06				Knowledge for the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: **reactor SCRAM**	3.8	16
295031 Reactor Low Water Level / 2	01						Knowledge of the operational implications of the following concepts as they apply to REACTOR LOW WATER LEVEL: **adequate core cooling**	4.7	17
295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown / 1	07						Knowledge of operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: **shutdown margin**	3.8	18
295038 High Offsite Release Rate / 9						2.1.27	Knowledge of system purpose and/or function	2.9	19
600000 Plant Fire On Site / 8					04		Ability to determine and/or interpret the following as they apply to PLANT FIRE ON SITE**: the fire's extent of potential operational damage to plant equipment**	3.1	20
295006 SCRAM / 1						2.4.16	Knowledge of EOP implementation hierarchy and coordination with other support procedures.	*4.0	*1
295016 Control Room Abandonment / 7					02		Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT: **reactor water level**	*4.3	*2
295021 Loss of Shutdown Cooling / 4					03		Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING: **reactor water level**	*3.5	*3
295024 High Drywell Pressure / 5						2.3.11	Ability to control radiation releases	*3.2	*4
295025 High Reactor Pressure / 3					03		Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: **suppression pool temperature**	*4.1	*5
295027 High Containment Temperature (Mark III) / 5					01		Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III): **containment temperature:**	*3.7	*6

							Mark III		
295031 Reactor Low Water Level / 2					01		Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: **reactor water level**	*4.6	*7
PAGE 2 TOTAL TIER 1 GROUP 1	3	0	1	0	6	3	PAGE TOTAL # QUESTIONS	13	
PAGE 1 TOTAL TIER 1 GROUP 1	3	2	3	4	2	0	PAGE TOTAL # QUESTIONS	14	
TIER 1 GROUP 1 TOTALS	6	2	4	4	8	3		27	

E/APE #/NAME/SAFETY FUNCTION	K1	K2	K3	A1	A2	G	TOPIC(S)	IMP	#
295002 Loss of Main Condenser Vacuum / 3									
295007 High Reactor Pressure / 3									
295008 High Reactor Water Level / 2		02					Knowledge of the interrelations between HIGH REACTOR WATER LEVEL and the following: **reactor feedwater system**	3.8	21
295009 Low Reactor Water Level / 2				03			Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL**:** **recirculation system (plant specific)**	3.1	22
295010 High Drywell Pressure/ 5		03					Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: **drywell/containment differential pressure**	3.1	23
295011 High Containment Temperature / 5					02		Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (MARK III)**: containment pressure**	4.1	24
295012 High Drywell Temperature / 5									
295013 High Suppression Pool Water Temp. / 5		01					Knowledge of the interrelations between HIGH SUPPRESSION POOL TEMPERATURE and the following: **suppression pool cooling**	3.7	25
295014 Inadvertent Reactivity Addition / 1					04		Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION: **violation of fuel thermal limits**	*4.4	*8
295015 Incomplete SCRAM / 1									
295017 High Offsite Release Rate / 9					05		Ability to determine and/or interpret the following as they apply to HIGH OFFSITE RELEASE RATE: **meteorological data**	*3.8	*9
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1			01				Knowledge of the reasons for the following responses as they apply to LOSS OF CRD PUMPS: **Reactor SCRAM**	3.9	26
295029 High Suppression									

Pool Water Level / 5									
295032 High Secondary Containment Area Temperature/ 5						2.4.5	Knowledge of the organization of the operating procedures network for normal / abnormal / and emergency evolutions.	3.6	27
295033 High Secondary Containment Area Radiation Levels / 9						2.1.25	Ability to obtain and interpret station reference material such as graphs / monographs / and tables which contain performance data.	*3.1	*10
295034 Secondary Containment Ventilation High Radiation / 9									
PAGE 1 TOTAL TIER 1 GROUP 2	0	3	1	1	3	2	PAGE TOTAL # QUESTIONS	10	

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**	**#**
295035 Secondary Containment High Differential Pressure / 5									
295036 Secondary Containment High Sump/Area Water Level / 5									
500000 High CTMT Hydrogen Conc. / 5									
PAGE 2 TOTAL TIER 1 GROUP 2	0	0	0	0	0	0	PAGE TOTAL # QUESTIONS	0	
PAGE 1 TOTAL TIER 1 GROUP 2	0	3	1	1	3	2	PAGE TOTAL # QUESTIONS	10	
TIER 1 GROUP 2 TOTALS	0	3	1	1	3	2	TIER 1 GROUP 2 TOTAL # QUESTIONS	10	
TIER 1 GROUP 1 TOTALS	6	2	4	4	8	3	TIER 1 GROUP 1 TOTAL # QUESTIONS	27	
TIER 1 TOTALS	6	5	5	5	11	5	TIER 1 TOTAL # QUESTIONS	37	

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	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
203000 RHR/LPCI: Injection Mode								16				Ability to **(a)** predict the impacts of the following on the RHR/LPCI: INJECTION MODE; and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **LOCA**	4.5	28
203000 RHR/LPCI: Injection Mode										04		Ability to manually operate and/or monitor in the control room: **heat exchanger cooling flow**	3.6	29
205000 Shutdown Cooling										04		Ability to manually operate and/or monitor in the control room: **heat exchanger cooling water valves**	3.3	30
206000 HPCI												N/A GGNS		
207000 Isolation (Emergency) Condenser												N/A GGNS		
209001 LPCS											2.1.33	Ability to recognize indications for system operating parameters which are entry-level conditions for Technical specifications.	4.0	31
209002 HPCS							02					Ability to predict and/or monitor changes in parameters associated with operating HPCS controls including: **HPCS pressure**	3.6	32
211000 SLC										06		Ability to manually operate and/or monitor in the control room: **RWCU system isolation**	3.9	33
212000 RPS			11									Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on the following: **recirculation system**	3.3	34
215003 IRM		01										Knowledge of electrical power supplies to the following: **IRM channels/detectors**	2.7	35
215003 IRM											2.2.11	Knowledge of the process for controlling temporary changes	3.4	36
215004 Source Range Monitor		01										Knowledge of electrical power supplies to the following: **SRM channels/detectors**	2.8	37
215005 APRM /												Knowledge of AVERAGE		38

LPRM				02								POWER RANGE MONITOR / LOCAL POWER RANGE MONITOR SYSTEM design features and/or interlocks which provide for the following: **reactor SCRAM signals**	4.2	
217000 RCIC										10		Ability to manually operate and/or monitor in the control room: **RCIC lights and alarms**	3.5	39
PAGE 1 TOTAL TIER 2 GROUP 1	0	2	1	1	0	0	1	1	0	4	2	PAGE 1 TIER 2 GROUP 1 TOTAL # QUESTIONS	12	

SYSTEM #/NAME	K 1	K 2	K 3	K 4	K 5	K 6	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
218000 ADS									02			Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: **ADS valve tailpipe temperatures**	3.7	40
218000 ADS	03											Knowledge of the physical connections and/or cause-effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following: **Nuclear boiler instrumentation system**	3.8	41
223002 PCIS / Nuclear Steam Supply Shutoff								04				Ability to **(a)** predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUT-OFF; and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **process radiation monitoring system failures**	3.2	42
223002 PCIS / Nuclear Steam Supply Shutoff						06						Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM / NUCLEAR STEAM SUPPLY SHUT-OFF: **various process instrumentation**	2.9	43
239002 SRVs			01									Knowledge of the effect that a loss or malfunction of the RELIEF/SAFETY VALVES will have on the following: **reactor pressure control**	4.0	44
239002 SRVs					02							Knowledge of implications of the following concepts as they apply to RELIEF/SAFETY VALVES: **safety function of SRV operation**	3.8	45
259002 Reactor Water Level Control							01					Ability to predict and/or monitor changes in parameters associated with operating REACTOR	3.8	46

												WATER LEVEL CONTROL SYSTEM controls including: **reactor water level**		
261000 SGTS	01											Knowledge of physical connections and/or cause-effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: **reactor building ventilation system**	3.6	47
262001 AC Electrical Distribution									01			Ability to monitor automatic operations of the AC ELECTRICAL DISTRIBUTION including: **breaker tripping**	3.2	48
262002 UPS (AC/DC)				01								Knowledge of UNINTERRUPTIBLE POWER SUPPLY (AC/DC) design features and/or interlocks which provide for the following: **transfer from preferred power to alternate power supplies**	3.4	49
PAGE 2 TOTAL TIER 2 GROUP 1	2	0	1	1	1	1	1	1	2	0	0	PAGE 2 TIER 2 GROUP 1 TOTAL # QUESTIONS	10	

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	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
263000 DC Electrical Distribution		01										Knowledge of electrical power supplies to the following: **major DC loads**	3.4	50
264000 EDGs					06							Knowledge of implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL): **load sequencing**	3.5	51
300000 Instrument Air	04											Knowledge of physical connections and/or cause-effect relationships between INSTRUMENT AIR SYSTEM and the following: **cooling water to compressor**	2.9	52
400000 Component Cooling Water						01						Knowledge of the effect that a loss or malfunction of the following will have on the CCWS: **valves**	2.8	53
205000 Shutdown Cooling											2.2.25	Knowledge of the bases in Technical Specifications for limiting conditions for operations	*3.7	*11

												and safety limits.		
212000 RPS								15				Ability to **(a)** predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **load rejection**	*3.8	*12
263000 DC Electrical Distribution								02				Ability to **(a)** predict the impacts of the following on the DC ELECTRICAL DISTRIBUTION; and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **loss of ventilation during charging**	*2.9	*13
264000 EDGs								05				Ability to **(a)** predict the impacts of the following on the EMERGENCY GENERATORS (DIESEL); and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **synchronization of the emergency generator with other electrical supplies**	*3.6	*14
300000 Instrument Air											2.2.14	Knowledge of the process for making configuration changes.	*3.0	*15
PAGE 3 TOTAL TIER 2 GROUP 1	1	1	0	0	1	1	0	3	0	0	2	PAGE 3 TIER 2 GROUP 1 TOTAL # QUESTIONS	9	
PAGE 1 TOTAL TIER 2 GROUP 1	0	2	1	1	0	0	1	1	0	4	2	PAGE 1 TIER 2 GROUP 1 TOTAL # QUESTIONS	12	
PAGE 2 TOTAL TIER 2 GROUP 1	2	0	1	1	1	1	1	1	2	0	0	PAGE 2 TIER 2 GROUP 1 TOTAL # QUESTIONS	10	
TOTAL TIER 2 GROUP 1	3	3	2	2	2	2	2	5	2	4	4	TIER 2 GROUP 1 TOTAL # QUESTIONS	31	

SYSTEM #/NAME	K 1	K 2	K 3	K 4	K 5	K 6	A1	A2	A3	A4	G	TOPIC(S)	IMP	#
201001 CRD Hydraulic														
201002 RMCS												N/A GGNS		
201003 Control Rod and Drive Mechanism			01									Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on the following: **reactor power**	3.4	54
201004 RSCS												N/A GGNS		
201005 RCIS														
201006 RWM												N/A GGNS		
202001 Recirculation														
202002 Recirculation Flow Control CFR41.6					02							Knowledge of implications of the following concepts as they apply to RECIRCULATION FLOW CONTROL SYSTEM: **feedback signals**	2.6	55
204000 RWCU														
214000 RPIS												N/A GGNS		
215001 Traversing In-Core Probe														
215002 RBM												N/A GGNS		
216000 Nuclear Boiler Instrumentation										03		Ability to manually operate and/or monitor in the control room: **process computer**	3.1	56
219000 RHR /LPCI Suppression Pool Cooling Mode								11				Ability to **(a)** predict the impacts of the following on the RHR/LPCI SUPPRESSION POOL COOLING MODE; <u>and</u> **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **motor operated valve failures**	3.3	57
223001 Primary CTMT and Auxiliaries											2.4.9	Knowledge of low power / shutdown implications in accident (LOCA or loss of RHR)	3.9	58

												mitigation strategies.		
226001 RHR/LPCI: CTMT Spray Mode														
230000 RHR/LPCI: Torus/Pool Spray Mode												N/A GGNS		
233000 Fuel Pool Cooling and Cleanup											2.1.2	Knowledge of operator responsibilities during all modes of operation.	4.0	59
234000 Fuel Handling Equipment								01				Ability to **(a)** predict the impacts of the following on the FUEL HANDLING EQUIPMENT; and **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **interlock failure**	*3.7	*16
PAGE 1 TOTAL TIER 2 GROUP 2	0	0	1	0	1	0	0	2	0	1	2	PAGE 1 TIER 2 GROUP 2 TOTAL # QUESTIONS	7	

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**	**A1**	**A2**	**A3**	**A4**	**G**	**TOPIC(S)**	**IMP**	**#**
239001 Main and Reheat Steam											2.4.6	Knowledge of symptom based EOP mitigation strategies.	*4.0	*17
239003 MSIV Leakage Control														
241000 Reactor/Turbine Pressure Regulator	14											Knowledge of physical connections and/or cause-effect relationships between REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following: **AC electrical power**	2.9	60
245000 Main Turbine Gen./Aux.														
256000 Reactor Condensate								13				Ability to **(a)** predict the impacts of the following on the REACTOR CONDENSATE SYSTEM; <u>and</u> **(b)** based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: **loss of applicable plant air systems**	*3.0	*18
259001 Reactor Feedwater														
268000 Radwaste										01		Ability to manually operate and/or monitor in the control room: **sump integrators**	3.6	61
271000 Offgas														
272000 Radiation Monitoring														
286000 Fire Protection			03									Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on the following: **plant protection**	3.8	62
288000 Plant Ventilation				02								Knowledge of PLANT VENTILATION SYSTEMS design features and/or interlocks	3.8	63

												which provide for the following: **secondary containment isolation**		
290001 Secondary CTMT														
290003 Control Room HVAC						02						Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROOM HVAC: **component cooling water systems**	2.9	64
290002 Reactor Vessel Internals											2.2.22	Knowledge of limiting conditions for operations and safety limits.	4.1	65
PAGE 2 TOTAL TIER 2 GROUP 2	1	0	1	1	0	1	0	1	0	1	2	PAGE 2 TIER 2 GROUP 2 TOTAL # QUESTIONS	8	

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	#
PAGE 1 TOTAL TIER 2 GROUP 2	0	0	1	0	1	0	0	2	0	1	2	PAGE 1 TIER 2 GROUP 2 TOTAL # QUESTIONS	7	
PAGE 2 TOTAL TIER 2 GROUP 2	1	0	1	1	0	1	0	1	0	1	2	PAGE 2 TIER 2 GROUP 2 TOTAL # QUESTIONS	8	
TOTAL TIER 2 GROUP 2	1	0	2	1	1	1	0	3	0	2	4	TIER 2 GROUP 2 TOTAL # QUESTIONS	15	
TOTAL TIER 2 GROUP 1	3	3	2	2	2	2	2	5	2	4	4	TIER 2 GROUP 1 TOTAL # QUESTIONS	31	
TOTAL TIER 2	4	3	4	3	3	3	2	8	2	6	8	TIER 2 TOTAL # QUESTIONS	46	

Facility: **Grand Gulf Nuclear Station** Date of Exam: **September 2007**

Category	K/ A#	Topic	RO IR	RO #	SRO-Only IR	SRO-Only #
1. Conduct Of Operations	2.1.32	Ability to explain and apply system limits and precautions.	3.8	66		
	2.1.10	Knowledge of conditions and limitations in the facility license.	3.9	67		
	2.1.8	Ability to coordinate personnel activities outside the control room.	3.6	68		
	2.1.14	Knowledge of system status criteria which require the notification of plant personnel.			*3.3	*19
	2.1.11	Knowledge of less than one hour Technical Specification action statements for systems.			*3.8	*20
	2.1					
	Subtotal			3		2
2. Equipment Control	2.2.2	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.	3.5	69		
	2.2.22	Knowledge of limiting conditions for operations and safety limits.	4.1	70		
	2.2.27	Knowledge of the refueling process.	3.5	71		
	2.2.19	Knowledge of maintenance work order requirements.			*3.1	*21
	2.2.21	Knowledge of pre and post maintenance operability requirements.			*3.5	*22
	2.2					
	Subtotal			3		2
3. Radiation Control	2.3.2	Knowledge of the facility ALARA program.	2.9	72		
	2.3.9	Knowledge of the process for performing a containment purge.	3.4	73		
	2.3.5	Knowledge of use and function of personnel monitoring equipment.			*2.5	*23
	2.3					
	2.3					
	2.3					
	Subtotal			2		1
4. Emergency Procedures /	2.4.46	Ability to verify that alarms are consistent with plant conditions.	3.6	74		
	2.4.4	Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal	4.3	75		

		operating procedures.				
Plan	2.4.30	Knowledge of which events related to system operations / status should be reported to outside agencies.			*3.6	*24
	2.4.28	Knowledge of procedures relating to emergency response to sabotage.			*3.3	*25
	2.4					
	2.4					
	Subtotal			2		2
Tier 3 Point Total				10		7

Facility: **Grand Gulf ~~Nuclear Station~~** Date of Examination: **21 May 2007** 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 w/ FCV fails to full open.	S; N; A	1
b. 204000 Reactor Water Cleanup System – Align RWCU for Vessel Level Control.	S; M; A; L	2
c. 241000 Reactor/Turbine Pressure Regulating System – Lower Reactor Pressure with Bypass Valves	S; M; A	3
d. 205000 Shutdown Cooling System (RHR Shutdown Cooling Mode) – Startup RHR Shutdown Cooling Mode.	S; D; L	4
e. 223001 Primary Containment System and Auxiliaries – Raise Suppression Pool Level using RCIC/HPCS	S; N; A	5 ESF
f. 262001 AC Electrical Distribution – Split BOP/ESF Loads followed by loss of transformer.	S; M; A	6 ESF
g. 286000 Fire Protection System – Perform the Control Room Actions in response to a Fire in the Auxiliary Building with a loss of Instrument Air.	C; N	8
h. N/A		
In-Plant Systems@ (3 for RO; 3 for SRO-I; 3or2 for SRO-U)		
i. 212000 Reactor Protection System – Startup RPS Motor Generator and Transfer RPS power from Alternate to Normal.	D	7
j. 201001 Control Rod Hydraulic System – *Rotate CRD Flow Control Valves*	R; D	1
k. 295016 Control Room Abandonment – Start SSW A & B and supply loads from Remote Shutdown Panel	N; E; L	*8* 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: **21 May 2007**
Exam Level (circle one) RO / SRO-I / **SRO-U** Operating Test Number:

System / JPM Title	Type Code*	Safety Function
Control Room Systems@ (8 for RO; 7 for SRO-I; 2 or 3 for SRO-U, including 1 ESF)		
a. 202001 Recirculation System – Startup idle Recirculation Pump w/ FCV fails to full open.	S; N; A	4
b. N/A		
c. N/A		
d. N/A		
e. 223001 Primary Containment System and Auxiliaries – Raise Suppression Pool Level using RCIC/HPCS	S; N; A	5 ESF
f. 262001 AC Electrical Distribution – Split BOP/ESF Loads followed by loss of transformer.	S; M; A	6 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. 201001 Control Rod Hydraulic System – *Rotate CRD Flow Control Valves*	R; N	1
k. 295016 Control Room Abandonment – Start SSW A & B and supply loads from Remote Shutdown Panel	N; E; L	*8* 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: **21 May 2007**	
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 plant conditions and plant personnel, determine staffing to meet shift requirements. GJPM-SRO-ADM-11 K/A 2.1.4: 3.4; 2.1.5: 3.4
Conduct of Operations	M	Given plant conditions, determine Plant Safety Index. (EOOS Factor) GJPM-SRO-ADM-12 K/A 2.1.19: 3.0
Equipment Control	M	Perform Operations Supervisor Review of Protective Tag out Clearance. GJPM-SRO-ADM-13 K/A 2.2.13: 3.8; 2.2.17: 3.5
Radiation Control	M	Review Liquid Radwaste Discharge Permit. GJPM-SRO-ADM-14 K/A 2.3.6: 3.1
Emergency Plan	M	Given plant conditions, determine entry into the Site Emergency Plan and complete the initial notification forms. Dry Fuel Storage GJPM-SRO-A&E-43 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

Safety Function/Knowledge & Ability/10CFR 55.45 Cross Reference

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **1** Op-Test No.: **Day 0**

Examiners: ______________________ Operators: ______________________

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

1. Startup 2nd RFPT and place on Master Controller.
2. Raise Reactor Power by withdrawing control rods.
3. *Respond to failed Reed Switch requiring substitute position.*
4. Respond to single control rod stuck per ONEP 05-1-02-IV-1.
5. Respond to APRM D failure upscale.
6. Respond to Pressure Controller fault Reactor Pressure rising.
7. Take actions per the EOPs in response to an ATWS and mitigate the consequences of the ATWS.
8. Respond to failure of Main Steam Bypass Valves to fully function.
9. *Respond to a failure of SLC to function properly.*

Initial Conditions: Reactor Power is at *50* %. 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

SRM 'F' are INOP and bypassed.
IRMs 'A' & 'H' are INOP and bypassed.
APRM 'H' is INOP due to failed downscale and is bypassed.
HPCS Pump is tagged out of service for failure of the Jockey Pump.
ESF 12 Transformer is tagged out of service for maintenance.
RPS 'A' is on Alternate Power due to EPA circuit breaker failure.
SBGT 'A' is operating for surveillance.

Appropriate clearances and LCOs are written.

Turnover: Continue plant startup per IOI-2. Ready for startup of RFPT 'B'. There are scattered thunder showers reported in the Tensas Parish area.

Scenario **1** Day **0** (Continued)

Event No.	Safety function	K/A[1]	10CFR 55.45[2]	Event Type*	Event Description
1	2	259001 A4.02: 3.9/3.7 A4.04: 3.1/2.9 A4.05: 4.0/3.9 A4.07: 3.3/3.2	4, 5, 8	N (RO)	Place RFPT B in service on the Master Level Controller. (SOI 04-1-01-N21-1)
2	1; 7	201005 A3.01: 3.5/3.5 A3.02: 3.5/3.5 A3.03: 3.4/3.3 A3.04: 3.3/3.3 A4.01: 3.7/3.7 A4.02: 3.7/3.7	1, 2, 3, 4, 5	R (RO)	Raise reactor power by using control rods to *52%*. (Control Rod Movement Sheet)
3	7	201005 A2.02: 2.8/3.2 A2.03: 3.2/3.2 A2.04: 3.2/3.2 A4.01: 3.7/3.7	3; 4; 5; 6	I (RO)	*Respond to a failed Reed Switch on control rod being moved requiring substitute position to be entered* *(SOI 04-1-01-C11-2)*
4	1	201005 A3.01: 3.5/3.5 A3.02: 3.5/3.5 A3.03: 3.4/3.3 A3.04: 3.3/3.3 A4.01: 3.7/3.7 A4.02: 3.7/3.7 201001 A4.04: 3.1/3.0 G2.1.12: 4.0 (SS) 201003 A2.01: 3.4/3.6 A3.01: 3.7/3.6	4, 5, 6	C (RO/ BOP) TS (SS)	Respond to a stuck control rod during withdrawal. (ONEP 05-1-02-IV-1) Complete **Technical Specification** determination.
5	7	215005 A4.05: 3.4/3.4 A3.07: 3.8/3.8 A3.08: 3.7/3.6 A2.04: 3.8/3.9 A2.02: 3.6/3.7 G2.1.12: 4.0 (SS)	3; 4	I (RO) TS (SS)	Respond to failure of APRM D upscale. Complete **Technical Specification** determination. (ARI 04-1-02-1H13-P680 7A-B11)

Scenario **1** Day **0** (Continued)

Event No.	Safety function	K/A[1]	10CFR 55.45[2]	Event Type*	Event Description
6	3	241000 A1.08: 3.3/3.2 A1.09: 3.3/3.3 A2.02: 3.7/3.7 A2.04: 3.7/3.8 A2.03: 4.1/4.2 A2.19: 3.8/3.8 A3.08: 3.8/3.8 A3.09: 3.3/3.2 A3.10: 3.3/3.3 A4.06: 3.9/3.9 A4.01: 3.9/4.0 295007 AA1.05: 3.7/3.8 AA2.01: 4.1/4.1	3; 4; 5; 7	C (RO)	Respond to a failure of the Reactor Pressure Control System with pressure rising. (ARI 04-1-02-1H13-P680 9A-D2)
7	1; 7	295006 AA1.01: 4.2/4.2 AA2.01: 4.5/4.6 AA2.02: 4.3/4.4 AA2.04: 4.1/4.1 AA2.05: 4.6/4.6 295015 AA1.02: 4.0/4.2 AA2.01: 4.1/4.3 AA2.02: 4.1/4.2 212000 A2.21: 3.6/3.9 A4.01: 4.6/4.6 A4.05: 4.3/4.3 A4.06: 4.2/4.1 A4.07: 4.0/3.9 A4.08: 3.4/3.4 A4.09: 3.9/3.8 A4.11: 3.7/3.7 A4.12: 3.9/3.9 A4.14: 3.8/3.8 A4.16: 4.4/4.4 A4.17: 4.1/4.1 295037 EA1.01: 4.6/4.6 EA1.03: 4.1/4.1 EA1.04: 4.5/4.5 EA1.05: 3.9/4.0 EA1.08: 3.6/3.6 EA2 ALL	3, 6	C (RO)	Recognize a failure to scram using RPS and manually scram the reactor using ATWS ARI. Respond to ATWS with partial Main Steam Bypass Valve availability. (EOP 05-1-01-EP-2A)

Scenario **1** Day **0** (Continued)

Event No.	Safety function	K/A[1]	10CFR 55.45[2]	Event Type*	Event Description
8	1	211000 A1 ALL A3 ALL A4.02: 4.2/4.2 A4.08: 4.2/4.2 295037 EA1.04: 4.5/4.5 EA1.10: 3.7/3.9	4; 5; 6	C (BOP)	Respond to a failure of Standby Liquid Control to initiate. (SOI 04-1-01-C41-1 and EOP 05-1-01-EP-2A)

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

[1] K/A G2.1.2: 3.0/4.0 Operator Responsibilities; G2.1.17: 3.5/3.6 Communication; G2.1.19: 3.0/3.0 Plant computer information for system status determination; G2.1.20: 4.3/4.2 Execute Procedural Steps; G2.1.31: 4.2/3.9 Locate and determine correct alignment of Control Room indications are covered during each evolution during the dynamic simulator examination.

[2] 10 CFR 55.45 (a) (3), (4), (12) and (13) are performed during each evolution during the dynamic simulator examination.

Critical Tasks

- Inserts rods by manual scrams and normal rod insertion using Attachments 18, 19, and 20.

- When allowed by Level / Power Control leg of EP-2A, restores injection from Condensate / Feedwater.

- Terminates and prevents all injection except boron, CRD and RCIC when required by steps L-7 or 8 of EP-2A.

  - For ATWS above 4% power, injects SLC A/B before Suppression Pool temperature reaches 110 degrees F.

Safety Function/Knowledge & Ability/10CFR 55.45 Cross Reference

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **2** Op-Test No.: **Day 0**

Examiners: ______________________ Operators: ______________________

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

1. Shift Main Turbine EHC Pumps.
2. Start RCIC for EPI and respond to RCIC over speed trip.
3. Raise Reactor Power using Reactor Recirc Flow.
4. Respond to a Main Steam Line Radiation Monitor failure downscale.
5. Respond to control rod drifting inward.
6. Respond to a LPRM failure downscale.
7. Respond to a loss of Main Condenser Vacuum.
8. Respond to a loss of Offsite Power with failure of Division 3 Diesel Generator.

Initial Conditions: Reactor Power is at 80 %.

INOPERABLE Equipment

SRM 'F' are INOP and bypassed.
IRMs 'A' & 'H' are INOP and bypassed.

Appropriate clearances and LCOs are written.

Turnover: Shift Main Turbine EHC pumps to 'A' and 'C' operating and 'B' in Standby. Once shifted raise Reactor Power to 90%. There are scattered thunder showers reported in the Tensas Parish area.

Event No.	Safety function	K/A[1]	10CFR 55.45[2]	Event Type*	Event Description
1	3	241000 A4.10: 2.9/2.9 G2.1.30: 3.9/3.4	6; 8	N (RO)	Shift operating EHC pumps. (SOI 04-1-01-N32-1)
2	1/4	202001 A4.04: 3.7/3.7 A4.02:3.5/3.4 202002 A4.04: 3.8/3.8 A4.08: 3.3/3.3	2; 6; 8	R (RO)	Raise Reactor Power using Recirc Flow to 90%.
3	2/4	217000 A4.01: 3.7/3.7 A4.03: 3.4/3.3 A2.02: 3.8/3.7	6; 8	C (BOP) TS	Start RCIC per EPI 04-1-03-E51-2 then respond to RCIC Overspeed Trip. Complete **Technical Specification**

		A1.05: 3.7/3.7		(SS)	determination. (Tech Spec 3.5.3)

Scenario **2** Day **0** (Continued)

Event No.	Safety function	K/A[1]	10CFR 55.45[2]	Event Type*	Event Description
4	7/9	272000 A2.16: 2.7/2.9 A2.06: 2.8/2.9 G2.2.22: 3.4/4.1	3; 9	TS (SS)	Respond to Main Steam Line Radiation Monitor failure downscale. (ARI 04-1-02-1H13-P601) Complete **Technical Specification** determination.(TR 3.3.6.1 Table TR 3.3.6.1-2)
5	1	202002 A2.13: 3.8/3.8 G2.2.22: 3.4/4.1	2; 4; 6; 8	I (RO) TS (SS)	Respond to control rod drifting inward. (ONEP 05-1-02-IV-1) Complete **Technical Specification** determination.(TS 3.1.3)
6	7	215005 a2.02: 3.6/3.7	4	I (RO) TS (SS)	Respond to a LPRM failure downscale. (ARI 04-1-02-1H13-P680; 17-S-02-40) Complete **Technical Specification** determination.(TS 3.3.1)
7	3	239001 A2.08: 3.6/3.6	4; 6; 8	C (ALL)	Respond to a loss of Main Condenser Vacuum. (ONEP 05-1-02-V-8)
8	6	262001 A2.03: 3.9/4.3	4; 6; 8	M (ALL)	Respond to a Loss of Offsite Power. (ONEP 05-1-02-I-4)
	6	264000 A2.09: 3.7/4.1	4; 6; 8		Respond to a failure of Division 3 Diesel Generator operate.

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

[1] K/A G2.1.2: 3.0/4.0 Operator Responsibilities; G2.1.17: 3.5/3.6 Communication; G2.1.19: 3.0/3.0 Plant computer information for system status determination; G2.1.20: 4.3/4.2 Execute Procedural Steps; G2.1.31: 4.2/3.9 Locate and determine correct alignment of Control Room indications are covered during each evolution during the dynamic simulator examination.

[2] 10 CFR 55.45 (a) (3), (4), (12) and (13) are performed during each evolution during the dynamic simulator examination.

Critical Tasks

- When level drops to <-191 inches or after level drops between TAF and – 191 inches, opens at least seven SRVs before level drops to – 212 inches. Pumps must be running and lined up for injection before reactor pressure drops to 300 psig.

Safety Function/Knowledge & Ability/10CFR 55.45 Cross Reference

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **3** Op-Test No.: **Day 0**

Examiners: ____________________ Operators: ____________________

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

1. Insert control rods to lower reactor power per control rod movement plan.
2. Respond to a failure of RPS MG 'B' per Loss of One or Both RPS Busses ONEP.
3. Respond to trip of SBGT 'A'.
4. Downshift Reactor Recirc Pumps to Slow Speed.
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 steam leak on RCIC when initiated.

Initial Conditions: Reactor Power is at *53* %. Plant shutdown is in progress in preparation for an outage. Reactor Recirculation pumps in Fast Speed; a single Reactor Feed Pump in Three element Master Level Control; one Heater Drain Pump is pumping forward.

INOPERABLE Equipment

SRM 'F' are INOP and bypassed.
IRMs 'A' & 'H' are INOP and bypassed.
APRM 'H' is INOP due to a failed FCTR card.
HPCS Pump is tagged out of service for failure of the Jockey Pump.
ESF 12 Transformer is tagged out of service for maintenance.
RPS 'A' is on Alternate Power due to EPA circuit breaker failure.
SBGT 'A' is operating for surveillance.

Appropriate clearances and LCOs are written.

Turnover: Continue plant shutdown per IOI-2. There are scattered thunder showers reported in the Tensas Parish area.

Scenario **3** Day **0** (Continued)

Event No.	Safety function	K/A[1]	10CFR 55.45[2]	Event Type*	Event Description
1	1; 7	201005 A3.01: 3.5/3.5 A3.02: 3.5/3.5 A3.03: 3.4/3.3 A3.04: 3.3/3.3 A4.01: 3.7/3.7 A4.02: 3.7/3.7	1, 2, 3, 4, 5	R (RO)	*Lower Reactor power using control rods to 60 – 75% rod line. (Control Rod Movement Sequence)*
2	1; 4	202001 A3.02: 3.1/3.0 A3.05: 2.9/2.9 A4.01: 3.7/3.7 A4.02: 3.5/3.4	3; 4; 5; 7	N (RO)	Transfer Reactor Recirc Pumps to Slow Speed (SOI 04-1-01-B33-1)
3	7	212000 A2.01: 3.7/3.9 A4.14: 3.8/3.8 G2.1.12: 4.0 (SS) G2.1.32: 3.8	3; 4, 5, 6	C (RO) TS (SS)	Respond to RPS Motor Generator 'B' trip. (ONEP 05-1-02-III-2). Complete **Technical Specification/FSAR** determination.
4	9	261000 A2.05: 3.0/3.1 G2.1.12: 4.0 (SS)	3; 4, 5, 6	C (BOP) TS (SS)	Respond to trip of SBGT 'A' trip. (ARI 04-1-02-1H13-P870 2A-A2) Complete **Technical Specification** determination.
5	2	259001 A3.03: 3.3/3.2 A3.06: 3.1/3.1 A4.01: 3.6/3.5 A4.02: 3.9/3.7 A4.04: 3.1/2.9 A4.07: 3.3/3.2 295009 AA1.01: 3.9/3.9 AA1.02: 4.0/4.0 AA2.01: 4.2/4.2 295031 EA1.02: 4.5/4.5 EA1.11: 4.1/4.1 EA1.12: 3.9/4.1 EA2. ALL	3; 4, 5, 6	M (ALL)	Respond to indications of large break LOCA on Feedwater Line 'A' per EOPs. (B21-F065A will close if attempted.)

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

Event No.	Safety function	K/A[1]	10CFR 55.45[2]	Event Type*	Event Description
	5	295032 EA1.01: 3.6/3.7 EA1.02: 3.4/3.5 EA1.05: 3.7/3.9 EA2.01: 3.8/3.8 EA2.03: 3.8/4.0	3; 4; 8	C (BOP)	RCIC steam leak will isolate by manual means.

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

[1] K/A G2.1.2: 3.0/4.0 Operator Responsibilities; G2.1.17: 3.5/3.6 Communication; G2.1.19: 3.0/3.0 Plant computer information for system status determination; G2.1.20: 4.3/4.2 Execute Procedural Steps; G2.1.31: 4.2/3.9 Locate and determine correct alignment of Control Room indications are covered during each evolution during the dynamic simulator examination.

[2] 10 CFR 55.45 (a) (3), (4), (12) and (13) are performed during each evolution during the dynamic simulator examination.

Critical Tasks

- Lower RPV Pressure to facilitate restoration of RPV Level using Condensate/Feedwater or ECCS.
- Isolate RCIC following EP-4 entry due to steam leak.