APRIL 2000 GGNS NRC EXAMINATION
JOB PERFORMANCE MEASURES

	TITLE
GG-1-JPM-RO-B3306	SECOND RECIRC PUMP STARTUP (POWER 30%)
GG-1-JPM-RO-G3301	ALIGN RWCU FOR VESSEL LEVEL CONTROL **(F)**
GG-1-JPM-RO-E3003	MANUAL INITIATION OF SPMU **(F)**
GG-1-JPM-RO-N3201	RAISE REACTOR PRESSURE AT 80 °F/HR
GG-1-JPM-RO-E1225	STARTUP ADHR IN REACTOR TO REACTOR MODE
GG-1-JPM-RO-T4801	PLACE SBGT TRAIN IN STANDBY WITH AUTO START **(F)**
GG-1-JPM-RO-EP030	DEFEAT RCIS PER EP 2 ATTACHMENT 20
GG-1-JPM-NLO-C1102	RECHARGE A HYDRAULIC CONTROL UNIT
GG-1-JPM-NLO-R2111	RACK OUT G. E. POWER VAC CIRCUIT BREAKER
GG-1-JPM-NLO-P6401	PERFORM LOCAL START OF DIESEL FIRE PUMP **(F)**

BACKUP JPMs

GG-1-JPM-RO-E5111	RCIC INITIATION
GG-1-JPM-RO-EP020	SCRAM RODS LOCALLY
GG-1-JPM-RO-EP004	DEFEAT HPCS HIGH SUPPRESSION POOL XFER

APRIL 2000 GGNS NRC EXAMINATION
ADMINISTRATIVE JOB PERFORMANCE MEASURES

GG-1-JPM-OP-ADM21	FUSE REMOVAL EFFECTS
GG-1-JPM-RO-ADM22	AC/DC SURVEILLANCE
GG-1-JPM-RO-ADM23	PREPARE TAGOUT
GG-1-JPM-OP-ADM24	ENTER CAA/CONTAMINATION AREA AND EXIT
GG-1-JPM-RO-ADM25	PERFORM INITIAL OHL NOTIFICATION
GG-1-JPM-SRO-ADM26	COMPLETE AN LCO
GG-1-JPM-SRO-ADM27	REVIEW A COMPLETED SURVEILLANCE
GG-1-JPM-SRO-A&E21	EAL CLASSIFICATION SCENARIO 1
GG-1-JPM-SRO-A&E22	EAL CLASSIFICATION SCENARIO 2
GG-1-JPM-SRO-A&E23	EAL CLASSIFICATION SCENARIO 3
GG-1-JPM-SRO-A&E24	EAL CLASSIFICATION SCENARIO 4 (BACKUP)
GG-1-JPM-SRO-A&E25	EAL CLASSIFICATION SCENARIO 5 (BACKUP)

Facility: **GRAND GULF NUCLEAR STATION** Date of Examination: **04/03/2000 – 04/06/2000**

	1. GENERAL CRITERIA	Initials a	Initials b	Initials c
a.	The operating test conforms with the previously approved outline; changes are consistent with sampling requirements (e.g., 10 CFR 55.45, operational importance, safety function distribution).			
b.	There is no day-to-day repetition between this and other operating tests to be administered during this examination.			
c.	The operating test shall not duplicate items from the applicants' audit test(s)(see Section D.1.a).			
d.	Overlap with the written examination and between operating test categories is within acceptable limits.			
e.	It appears that the operating test will differentiate between competent and less-than-competent applicants at the designated license level.			
	2. WALK-THROUGH (CATEGORY A & B) CRITERIA	--	--	--
a.	Each JPM includes the following, as applicable: + initial conditions + initiating cues + references and tools, including associated procedures + validated time limits (average time allowed for completion) and specific designation if deemed to be time critical by the facility licensee + specific performance criteria that include: - detailed expected actions with exact criteria and nomenclature - system response and other examiner cues - statements describing important observations to be made by the applicant - criteria for successful completion of the task - identification of critical steps and their associated performance standards - restrictions on the sequence of steps, if applicable			
b.	The prescripted questions in Category A are predominantly open reference and meet the criteria in Attachment 1 of ES-301.			
c.	Repetition from operating tests used during the previous licensing examination is within acceptable limits (30% for the walk-through) and do not compromise test integrity.			
d.	At least 20 percent of the JPMs on each test are new or significantly modified.			
	3. SIMULATOR (CATEGORY C) CRITERIA	--	--	--
a.	The associated simulator operating tests (scenario sets) have been reviewed in accordance with Form ES-301-4 and a copy is attached.			

Printed Name / Signature Date

a. Author _____

b. Facility Reviewer(*) _____

c. NRC Chief Examiner (*) _____

d. NRC Supervisor (*) _____

(*) The facility signature is not applicable for NRC-developed tests; two independent NRC reviews are required.

Facility: **GRAND GULF NUCLEAR STATION** Date of Exam: **04/03/2000 – 04/06/2000**

Scenario Numbers: **1** / **2** / **3**

QUALITATIVE ATTRIBUTES	Initials a	Initials b	Initials c
1. The initial conditions are realistic, in that some equipment and/or instrumentation may be out of service, but it does not cue the operators into expected events.			
2. The scenarios consist mostly of related events.			
3. Each event description consists of + the point in the scenario when it is to be initiated + the malfunction(s) that are entered to initiate the event + the symptoms/cues that will be visible to the crew + the expected operator actions (by shift position) + the event termination point (if applicable)			
4. No more than one non-mechanistic failure (e.g., pipe break) is incorporated into the scenario without a credible preceding incident such as a seismic event.			
5. The events are valid with regard to physics and thermodynamics.			
6. Sequencing and timing of events is reasonable, and allows the examination team to obtain complete evaluation results commensurate with the scenario objectives.			
7. If time compression techniques are used, the scenario summary clearly so indicates. Operators have sufficient time to carry out expected activities without undue time constraints. Cues are given.			
8. The simulator modeling is not altered.			
9. The scenarios have been validated. Any open simulator performance deficiencies have been evaluated to ensure that functional fidelity is maintained while running the planned scenarios.			
10. Every operator will be evaluated using at least one new or significantly modified scenario. All other scenarios have been altered in accordance with Section D.4 of ES-301.			
11. All individual operator competencies can be evaluated, as verified using Form ES-301-6 (submit the form along with the simulator scenarios).			
12. Each applicant will be significantly involved in the minimum number of transients and events specified on Form ES-301-5 (submit the form with the simulator scenarios).			
13. The level of difficulty is appropriate to support licensing decisions for each crew position.			

TARGET QUANTITATIVE ATTRIBUTES (PER SCENARIO; SEE SECTION D.4.D)	1 2 3 Actual Attributes	--	--	--
1. Total malfunctions (5-8)	**6 / 6 / 5**			
2. Malfunctions after EOP entry (1-2)	**3 / 2 / 2**			
3. Abnormal events (2-4)	**2 / 3 / 2**			
4. Major transients (1-2)	**1 / 1 / 1**			
5. EOPs entered/requiring substantive actions (1-2)	**2 / 2 / 2**			
6. EOP contingencies requiring substantive actions (0-2)	**1 / 1 / 1**			
7. Critical tasks (2-3)	**2 / 5 / 3**			

GRAND GULF NUCLEAR STATION **DATES: 04/03/2000 – 04/06/2000**

Applicant Type	Evolution Type	Minimum Number	Scenario Number			
			BOP 1	ATC 2	BOP 3	TOTAL
RO	Reactivity	1	0	1	0	1
	Normal	1	0	1	1	2
	Instrument	2	2	2	0	4
	Component	2	3	2	2	7
	Major	1	1	1	1	3
As RO	Reactivity	1				
	Normal	0				
	Instrument	1				
	Component	1				
	Major	1				
SRO-I						
As SRO	Reactivity	0				
	Normal	1				
	Instrument	1				
	Component	1				
	Major	1				
SRO-U	Reactivity	0				
	Normal	1				
	Instrument	1				
	Component	1				
	Major	1				

Instructions: (1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.4.d) but must be significant per Section C.2.a of Appendix D.

Author:

Chief Examiner:

ES-301 Competencies Checklist Form ES-301-6

GRAND GULF NUCLEAR STATION **DATES: 04/03/2000 – 04/06/2000**

Competencies	Applicant #1 RO/SRO-I/SRO-U			Applicant #2 RO/SRO-I/SRO-U			Applicant #3 RO/SRO-I/SRO-U		
	SCENARIO			SCENARIO			SCENARIO		
	BOP 1	ATC 2	BOP 3	SS 1	BOP 2	ATC 3	ATC 1	SS 2	SS 3
Understand and Interpret Annunciators and Alarms	4, 6	3, 4, 5, 7	4, 5	1, 3, 4, 5, 6, 7	5, 9	3	1, 2, 3, 4	3, 4, 5, 7, 9	3, 4
Diagnose Events and Conditions	4, 8, 9	4, 5, 7, 8	4, 5, 8	3, 4	5, 9	7	4, 7	4, 5, 7, 8, 9	3, 4, 5, 7, 8
Understand Plant and System Response	3, 4, 6, 7, 8	1, 2	2, 8	3, 4, 6, 7	7, 9	1, 7	1, 2, 3	1, 2, 7, 9	1, 3, 5, 6, 8
Comply With and Use Procedures (1)	3, 4, 6	1 – 8	2, 4	ALL	5, 7	1, 3, 6	1, 2, 3, 4	ALL	ALL
Operate Control Boards (2)	3, 6, 7, 8, 9	1, 2, 3, 6, 7, 8	2, 4, 5, 8	N/A (4)	7, 8, 9	1, 3, 6, 7, 8	1, 2, 5, 7	N/A (4)	N/A (4)
Communicate and Interact With the Crew	4, 6, 8, 9	1 - 8	2, 4, 5	ALL	5, 7, 9	1, 3	1, 2, 4, 5, 7	ALL	ALL
Demonstrate Supervisory Ability (3)	N/A	N/A	N/A	ALL	N/A	N/A	N/A	ALL	ALL
Comply With and Use Tech. Specs. (3)	N/A	N/A	N/A	3, 4	N/A	N/A	N/A	3	4

Notes:

(1) Includes Technical Specification compliance for an RO.
(2) Optional for an SRO-U.
(3) Only applicable to SROs.
(4) Only applicable to RO positions at GGNS.

Instructions:

Circle the applicant's license type and enter one or more event numbers that will allow the examiners to evaluate every applicable competency for every applicant.

Author: _____

Chief Examiner: _____

Facility: **GRAND GULF NUCLEAR STATION** Date of Exam: **04/03/2000 – 04/06/2000**

Scenario Numbers: ***BACKUP SCENARIOS* 1** / **2** (Scenarios 4 & 5)

QUALITATIVE ATTRIBUTES	Initials a	Initials b	Initials c
1. The initial conditions are realistic, in that some equipment and/or instrumentation may be out of service, but it does not cue the operators into expected events.			
2. The scenarios consist mostly of related events.			
3. Each event description consists of + the point in the scenario when it is to be initiated + the malfunction(s) that are entered to initiate the event + the symptoms/cues that will be visible to the crew + the expected operator actions (by shift position) + the event termination point (if applicable)			
4. No more than one non-mechanistic failure (e.g., pipe break) is incorporated into the scenario without a credible preceding incident such as a seismic event.			
5. The events are valid with regard to physics and thermodynamics.			
6. Sequencing and timing of events is reasonable, and allows the examination team to obtain complete evaluation results commensurate with the scenario objectives.			
7. If time compression techniques are used, the scenario summary clearly so indicates. Operators have sufficient time to carry out expected activities without undue time constraints. Cues are given.			
8. The simulator modeling is not altered.			
9. The scenarios have been validated. Any open simulator performance deficiencies have been evaluated to ensure that functional fidelity is maintained while running the planned scenarios.			
10. Every operator will be evaluated using at least one new or significantly modified scenario. All other scenarios have been altered in accordance with Section D.4 of ES-301.			
11. All individual operator competencies can be evaluated, as verified using Form ES-301-6 (submit the form along with the simulator scenarios).			
12. Each applicant will be significantly involved in the minimum number of transients and events specified on Form ES-301-5 (submit the form with the simulator scenarios).			
13. The level of difficulty is appropriate to support licensing decisions for each crew position.			

BACKUP SCENARIOS **TARGET QUANTITATIVE ATTRIBUTES (PER SCENARIO; SEE SECTION D.4.D)**	**4 5** Actual Attributes	--	--	--
1. Total malfunctions (5-8)	**7 / 5**			
2. Malfunctions after EOP entry (1-2)	**4 / 2**			
3. Abnormal events (2-4)	**2 / 3**			
4. Major transients (1-2)	**1 / 2**			
5. EOPs entered/requiring substantive actions (1-2)	**2 / 2**			
6. EOP contingencies requiring substantive actions (0-2)	**1 / 2**			
7. Critical tasks (2-3)	**3 / 5**			

GRAND GULF NUCLEAR STATION **DATES: 04/03/2000 – 04/06/2000**

Applicant Type	Evolution Type	Minimum Number	Scenario Number			
			SS 1	BOP 2	ATC 3	TOTAL
RO	Reactivity	1				
	Normal	1				
	Instrument	2				
	Component	2				
	Major	1				
As RO	Reactivity	1		0	1	1
	Normal	0		0	0	0
	Instrument	1		1	1	2
	Component	1		1	1	2
	Major	1		1	1	2
SRO-I 1&2						
As SRO	Reactivity	0	0			0
	Normal	1	1			1
	Instrument	1	2			2
	Component	1	3			3
	Major	1	1			1
SRO-U	Reactivity	0				
	Normal	1				
	Instrument	1				
	Component	1				
	Major	1				

Instructions: (1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(2) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.4.d) but must be significant per Section C.2.a of Appendix D.

Author:

GRAND GULF NUCLEAR STATION **DATES: 04/03/2000 – 04/06/2000**

BACK UP SCENARIOS

Applicant Type	Evolution Type	Minimum Number	Scenario Number			
			4	5		
RO ATC	Reactivity		1	1		
	Normal		0	1		
	Instrument		1	1		
	Component		1	0		
	Major		1	2		
As **RO BOP**	Reactivity		0	0		
	Normal		1	0		
	Instrument		1	0		
	Component		3	2		
	Major		1	2		
SRO-I						
As **SRO SS**	Reactivity		0	0		
	Normal		1	1		
	Instrument		2	1		
	Component		4	2		
	Major		1	3		
SRO-U	Reactivity					
	Normal					
	Instrument					
	Component					
	Major					

Instructions: (1) Enter the operating test number and Form ES-D-1 event numbers for each evolution type.

(3) Reactivity manipulations may be conducted under normal or *controlled* abnormal conditions (refer to Section D.4.d) but must be significant per Section C.2.a of Appendix D.

Author: _____

Chief Examiner: _____

GRAND GULF NUCLEAR STATION **DATES: 04/03/2000 – 04/06/2000**

BACKUP SCENARIOS	Applicant #1 **RO**/SRO-I/SRO-U		Applicant #2 RO/**SRO-I**/SRO-U		Applicant #3 RO/**SRO-I**/SRO-U	
Competencies	**SCENARIO**		**SCENARIO**		**SCENARIO**	
	ATC 4	**ATC 5**	**BOP 4**	**BOP 5**	**SS 4**	**SS 5**
Understand and Interpret Annunciators and Alarms	3, 4, 5	2, 5, 6	1, 6, 7, 8, 9	4, 7	1, 3, 4, 5, 6, 7, 8, 9	2, 4, 5, 6, 7
Diagnose Events and Conditions	3, 5	2, 5	5, 6, 7, 8, 9	4, 7	3, 4, 5, 6	2, 4, 5, 6, 7
Understand Plant and System Response	3, 4	5, 6	1, 5	4, 5, 7	3, 4, 5	2, 4, 5, 6, 7
Comply With and Use Procedures (1)	2, 3, 4, 5	1, 2, 3, 6	1, 6	5, 6, 7	ALL	ALL
Operate Control Boards (2)	2, 3, 4, 5	1, 2, 3, 5, 6	1, 6	4, 5, 6, 7	N/A (4)	N/A (4)
Communicate and Interact With the Crew	2, 3, 5	1, 2, 3, 5, 6	1, 6, 7, 8, 9	4, 5, 6, 7	ALL	ALL
Demonstrate Supervisory Ability (3)	N/A	N/A	N/A	N/A	ALL	ALL
Comply With and Use Tech. Specs. (3)	N/A	N/A	N/A	N/A	1	4

Notes:

(1) Includes Technical Specification compliance for an RO.
(2) Optional for an SRO-U.
(3) Only applicable to SROs.
(4) Only applicable to ROs at GGNS.

Instructions:

Circle the applicant's license type and enter one or more event numbers that will allow the examiners to evaluate every applicable competency for every applicant.

Author: _____

Chief Examiner: _____

JPMs

LOCATION	SYSTEM	SAFETY FUNCTION	PHYSICAL LOCATION
SIMULATOR	RECIRC	1	
	RWCU	2	**
	SPMU	5	
	PRESSURE CONTROL	3	
	ADHR	4	**
	SBGT	9	**
CONTROL ROOM	RCIS	7	Control Room
PLANT	CRD	1	Containment
	ELECTRICAL DIST	6	Old Training Building
	FIRE PROTECTION	8	Fire Water Pump House

**** JPMs could be done in the same simulator setup as required
BACKUP JPMs ARE AVAILABLE FOR ANY NEEDED.

ADMIN JPMS

ADMIN SECTION WILL BE DONE MONDAY AFTERNOON.

RO A.2 will be performed on a Computer having access to the Red Tag Program and having Prints and Procedures available.

A.3 Radiation Protection will be performed in the Radiation Worker Training Practical Facility.

SRO A.4 will be performed at the end of the Scenario which the SRO candidate is the Shift Supervisor.

RO A.4 will be performed in the Simulator.

I will set up two rooms in the EOF with controlled sets of procedures, Tech Specs, EOPs, and a full set of prints for evaluators to use for administering Admin JPMs. (OEC Office & Public Information Office in EOF)

I intend to utilize our EOF for maintaining candidates who are not being evaluated and they will have a facility monitor in the room for exam security. This room has rest room facilities and we will have food and drinks available in the room. This area is centrally located to the Simulator and the EOF Admin JPM rooms.

CANDIDATES WILL BE SEQUESTERED WHEN NOT IN THE CUSTODY OF AN EVALUATOR.

SCENARIOS

CREW 1

SCENARIO	RO 1	SRO 1	SRO 3
DAY 1 SCENARIO 1	BOP	SS	ATC
DAY 1 SCENARIO 2	ATC	BOP	SS
DAY 2 SCENARIO 3	BOP	ATC	SS

CREW 2

SCENARIO	RO 2	SRO 2	SRO 4
DAY 1 SCENARIO 1	BOP	SS	ATC
DAY 1 SCENARIO 2	ATC	BOP	SS
DAY 2 SCENARIO 3	BOP	ATC	SS

CREWS WILL BE SEQUESTERED WHEN NOT IN THE SIMULATOR.

Facility: **GRAND GULF NUCLEAR STATION** Date of Examination: **4/3/00 - 4/5/00** **ES-301-2**

Exam Level (circle one): ***BACKUP JPMS*** Operating Examination

System / JPM Title / Type Codes*	Safety Function	Knowledge / Ability	IMP.	Additional K/A's	ORIGIN	NOTES
1. 217000 RCIC **(N) (S) (A)** Manually initiate RCIC. (Failure of E51-F046 to open.) GG-1-JPM-RO-E5111	2 / 4 8 min.	A3.01	3.5 / 3.5	A2.08: 3.0/3.1 A4.04: 3.6/3.6	**BANK**	Manual Initiation. Pb works. ***Backup simulator.***
2. 212000 RPS **(D) (P) (R)** Scram Control Rods at the HCUs. GG-1-JPM-RO-EP020	7 / 1 13 min.	A4.17	4.1 / 4.1	295037 EK3.07: 4.2/4.3	**BANK**	EP-2 Att. 22 Emergency procedure ***Backup plant.***
3. 209002 HPCS **(D) (C)** Defeat HPCS High Suppression Pool Level Suction Transfer Interlock. GG-1-JPM-RO-EP004	2 / 5 5 min.	A2.12	3.3 / 3.5	295029 EA1.02: 3.1/3.1	**BANK**	EP-2 Att. 4 Emergency procedure ***Backup Control Room.***

Facility: **GRAND GULF NUCLEAR STATION** Date of Examination: **4/3/00 - 4/5/00**

Examination Level (circle one): ***RO*** / SRO Operating Test Number: __1___

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions	Knowledge / Ability	IMP	Additional K/A's	ORIGIN	NOTES
A.1	Plant Drawings	JPM GG-1-JPM-OP-ADM21 Given a component, determine the fuse(s) to be removed to de-energize the component and how the component will fail.	2.1.24	2.8		**NEW**	NRC 3/98 was to determine impact of failed relay
	Technical Specifications	JPM GG-1-JPM-RO-ADM22 Perform AC/DC surveillance following a failure of Division III Diesel Generator.	2.1.31	4.2	2.1.20: 4.3	**NEW**	
A.2	Protective Tagging Program	JPM GG-1-JPM-RO-ADM23 Using the Tagging Computer prepare clearance for a given job.	2.2.13	3.6	2.1.24: 2.8	**NEW**	NRC 3/98 was to restore
A.3	Contamination Control	JPM GG-1-JPM-OP-ADM24 Perform required actions to access the Controlled Access Area (CAA), don Protective clothing, enter a contamination area and exit/remove protective clothing.	2.3.4	2.5		**NEW**	Performed in Rad Worker Training Trailer NRC 3/98 was entry and exit CAA only. **CFR 55.45a (9), (10)**
A.4	Emergency Plan Assessment	JPM GG-1-JPM-OP-ADM25 Given initial notification form, perform notification of offsite agencies using the Operational Hotline.	2.4.43	2.8	2.4.39: 3.3 2.4.30: 3.6	**NEW**	**CFR 55.45a (11)**

Facility: **GRAND GULF NUCLEAR STATION** Date of Examination: **4/3/00 - 4/5/00**

Exam Level (circle one): **RO** / SRO(I) / SRO(U) Operating Test No.: ___1___

System / JPM Title / Type Codes*	Safety Function	Knowledge / Ability	IMP.	Additional K/A's	ORIGIN	NOTES
B.1. CONTROL ROOM SYSTEMS						
1. 202001 RECIRCULATION SYSTEM **(D)(S)** Startup Second Recirc Pump (Power 30 %) GG-1-JPM-RO-B3306	**1**	A4.01	3.7	A4.04: 3.7	**BANK**	**ALSO SAFETY FUNCTION 4**
2. 204000 REACTOR WATER CLEANUP SYSTEM **(N)(S)(A)(L)** Align RWCU for Vessel Level Control (Faulted) GG-1-JPM-RO-G3301	**2**	A1.01	3.1	A1.04: 2.8 A1.06: 2.8 A2.01: 3.2	**NEW**	Failure of G33-F234
3. 223001 PRIMARY CONTAINMENT SYSTEM **(N)(S)(A)** Manual Initiation of SPMU (Faulted) GG-1-JPM-RO-E3003	**5**	A2.11	3.6	A3.01: 3.4 2.1.30: 3.9 2.1.31: 4.2	**BANK**	Similar to JPM NRC 3/98
4. 241000 REACTOR/TURBINE PRESSURE REGULATING SYSTEM **(N)(S)** Raise Reactor Pressure at 80 °F/hr using Turbine Bypass Controls GG-1-JPM-RO-N3201	**3**	A4.02	4.1	A4.06: 3.9	**NEW**	
5. 295021 LOSS OF SHUTDOWN COOLING **(N)(S)(L)** Startup Alternate Decay Heat Removal in Reactor to Reactor mode GG-1-JPM-RO-E1225	**4**	AA1.04	3.7	205000 A4.01: 3.7 A4.02: 3.6 A4.03: 3.6 A4.09: 3.1	**NEW**	

Facility: **GRAND GULF NUCLEAR STATION** Date of Examination: **4/3/00 – 4/5/00**

Exam Level (circle one): **RO** / SRO(I) / SRO(U) Operating Test No.: ___1___

System / JPM Title / Type Codes*	Safety Function	Knowledge / Ability	IMP.	Additional K/A's	ORIGIN	NOTES
6. 261000 STANDBY GAS TREATMENT SYSTEM **(N)(S)(A)** Place SBGT Train in Standby with an Auto Start Signal present (Faulted) GG-1-JPM-RO-T4801	**9**	A4.03	3.0	A4.02: 3.1 A4.09: 2.7 A2.01: 2.9	**NEW**	**CFR 55.45a (9)**
7. 201005 ROD CONTROL AND INFORMATION SYSTEM **(N)(C)** Defeat Rod Control and Information System per EP2 Attachment 20. GG-1-JPM-RO-EP030	**7**	A2.03	3.2	A2.06: 3.2 295037 EA1.08: 3.6	**NEW**	**ALSO SAFETY FUNCTION 1**
B.2. FACILITY WALK-THROUGH						
8. 201001 CONTROL ROD DRIVE HYDRAULIC **(D)(P)(R)** Recharge a Hydraulic Control Unit (HCU) GG-1-JPM-NLO-C1102	**1**	A2.10	3.5	201003 A2.08: 3.8	**BANK**	**Containment**
9. 262001 A.C. ELECTRICAL DISTRIBUTION **(D)(P)** Rack out G.E. Power Vac Circuit Breaker GG-1-JPM-NLO-R2111	**6**	A4.01	3.4	A4.03: 3.2	**BANK**	Old Training Building Mockup
10. 286000 FIRE PROTECTION SYSTEM **(N)(P)(A)** Perform Local Start of Fire Pump Diesel after Failure to Start (Faulted) GG-1-JPM-NLO-P6401	**8**	A4.06	3.4		**NEW**	Fire Pump House **Emergency / Abnormal**

* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (P)lant, (R)CA

NRC EXAMINATION APRIL 2000
GRANDGULF NUCLEAR STATION
NPF 50-416
U.S. NUCLEAR REGULATORY COMMISSION
OPERATOR LICENSE EXAMINATION
FOR
GRAND GULF NUCLEAR STATION
APRIL 2000

PROCEDURES AND STANDARDS

TRAINING SECTION INSTRUCTION
14-S-02-22, NRC EXAMINATION PREPARATION AND ADMINISTRATION

10 CFR 55, OPERATOR LICENSES SECTIONS 41, 43, 45

NUREG 1021 REVISION 8, OPERATOR LICENSING EXAMINATION STANDARDS FOR POWER REACTORS

NUREG 1123 REVISION 2, KNOWLEDGE AND ABILITIES CATALOG FOR NUCLEAR POWER PLANT OPERATORS: BOILING WATER REACTORS

NRC EXAMINATION APRIL 2000
GRANDGULF NUCLEAR STATION
NPF 50-416

JUSTIFICATIONS FOR DELETIONS ON
WRITTEN EXAMINATION OUTLINE

SYSTEMS DELETED

201002 Reactor Manual Control System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

201004 Rod Sequence Control System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

201006 Rod Worth Minimizer System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

214000 Rod Position Information System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

215002 Rod Block Monitor System - This system is not incorporated into the BWR-6 design. The functions of this system are incorporated into the Rod Control and Information System.

206000 High Pressure Core Injection (HPCI) - This system is not incorporated into the BWR 6 design.

207000 Isolation (Emergency) Condenser - This system is not incorporated into the BWR 6 design. This was replaced by the Mark III Containment Suppression Pool.

219000 RHR/LPCI: Torus Cooling Mode - The BWR 6 Mark III Containment utilizes a Suppression Pool instead of a Torus.

230000 RHR/LPCI: Torus/Pool Spray Mode - This system is not incorporated into the BWR 6 Mark III Containment design.

NRC EXAMINATION APRIL 2000
GRANDGULF NUCLEAR STATION
NPF 50-416

WRITTEN EXAMINATION

Abilities that were hit on the random method of topic selection for the Generic Knowledge and Abilities that would be better examined in the Operating Examination were replaced with alternate random selections.

Knowledge and Abilities which were hit on the random selection which had Importance values < 2.5 were replaced with alternate random selections.

Knowledge and Abilities that were hit on the random selection which in NUREG 1123 listed NONE were replaced with alternate topics for the System or Evolution.

Plant Generics Knowledge and Abilities

Based on 127 topic areas that apply to a single unit BWR.

			RO	SRO
Conduct of Operations	34 topics	= 27%	4	5
Equipment Control	32 topics	= 25%	3(2)	4
Radiation Control	11 topics	= 9%	1(2)	2
Emergency Procedures/Plan	50 topics	= 39%	5	6
TOTAL	127	100%	13	17

BOTH WRITTEN

Topic 500000 originally selected as Emergency Knowledge 1. The only topic in EK1 was Containment integrity. Changed to generic 2.1.20 for execution of procedural steps concerning Hydrogen Recombiners.

Topic 295030 Moved from EA1 to generics based on number of questions in A1 and allowance for basis question on Suppression Pool low level.

Topic 295001 Moved from AA2 that was random to generics of AOP operations.

Topic 295002 Moved from AA1 that was random selection to AK3 to determine basis for MSIV Isolation on loss of condenser vacuum AA1 topics tend to be lower level knowledge.

Topics 295028 and 295012 both were selected to topic A1, 295028 was moved to A2 to allow different knowledge about High Drywell Temperature to be tested.

Topic 212000 random selection A3 moved to generics to encompass indication of failure which can cause an inadvertent partial reactor scram.

Topic 226001 random selection was K5, moved to A1 due to low discretionary value of topic 226001 K5.

Topic 201001 random selection K5 moved to A1 due to low importance factors of K5 topics and number of K5s selected on random.

Topic 300000 random selection was K2, K2 only had two topics which had no discretionary value. Replaced with K3.02 dealing with system interrelations.

Topic 263000 random selection was K4 that is N/A at GGNS, changed to K3.02.

Topic 271000 random selection was K3, changed to K4.04 to allow question concerning precautions about newly installed Hydrogen Water Chemistry System and hazards.

Topic 268000 random selection was K3, topic K3 does not have any discriminatory value, changed to generics 2.4.21.

Topic 280002 random selection K1 topics sufficiently covered in other areas of examination, moved to K5 to evaluate Brittle Fracture Limits.

SENIOR REACTOR OPERATOR WRITTEN

NONE as of 11/23/99

REACTOR OPERATOR WRITTEN

Topic 218000 random selection was A4 changed to A2, previous question on ADS sufficiently covered topic A4.

Topic 201005 random selection 223002 A3 was changed to 201005 Generics, due to A1 and A3 similarities of topic and A1 was previously used and a topic for reactivity management was needed.

Topic 201003 random selection was K6 that was covered on NRC exam 1998, changed to K4.

BWR RO EXAMINATION OUTLINE
EMERGENCY & ABNORMAL PLANT
EVOLUTIONS - TIER 1 GROUP 1

E/APE#/NAME/SAFETY FUNCTION	K 1	K 2	K 3	A 1	A 2	G	TOPIC(S)	IMP	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
295005 Main Turbine Generator Trip / III CFR41.4/41.5/41.6			02				Describe the basis for a Recirc Pump downshift at the end of cycle upon a Main Turbine Trip and when it is in effect.	3.4		**RO**	202001 K4.13: 3.7	NEW	
295006 SCRAM / I CFR41.5/41.14		02					Describe the response of the Reactor Water Level Control System when reactor level reaches level 3 during a reactor scram.	3.8		BOTH		NEW	
295007 High Reactor Pressure / III CFR41.7					02		With the reactor operating at 100% power, describe the effects on the reactor if a MSIV isolates.	4.1		BOTH	AA2.03: 3.7	BANK	Reactor power and reactor level inside shroud
295009 Low Reactor Water Level / II CFR41.2/41.14	05						Identify the Reactor level required to provide adequate core circulation with no recirculation pumps or shutdown cooling system operating.	3.3		BOTH		NEW	
295010 High Drywell Pressure / V CFR41.4/41.9	02						Identify the Drywell pressure that will begin to clear the Drywell to Containment Suppression Pool Vents.	2.8		BOTH		NEW	
295014 Inadvertent Reactivity Addition / I CFR41.1/41.2/41.6/43.6			01				With the reactor in startup conditions such that the reactor has dropped subcritical, what are the operator actions if a high worth control rod is withdrawn fully.	4.1		BOTH		NEW	Susquehanna reactivity event 7/98
295015 Incomplete SCRAM / I CFR41.6/43.6			01				Describe the method to be used to allow the insertion of control rods using RCIS during an ATWS.	3.4		BOTH		NEW	
295024 High Drywell Pressure / V CFR41.9	01						Describe the implications of exceeding the Drywell maximum design pressure.	4.1		BOTH		NEW	
295025 High Reactor Pressure / III CFR41.3/41.5/41.7/43.2		05					Describe the basis for the operation of Low-Low Set Safety Relief Valves.	4.1		BOTH		NEW	
295031 Reactor Low Water Level / II CFR41.7/41.10/43.5		12					Given plant conditions, determine the reaction of plant systems.	4.5		BOTH		NEW	
PAGE 1 TOTAL TIER 1 GROUP 1	3	3	3	0	1	0	PAGE TOTAL # QUESTIONS	10					

E/APE#/NAME/SAFETY FUNCTION	K 1	K 2	K 3	A 1	A 2	G	TOPIC(S)	IMP	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / I CFR41.1/41.2/41.6/43.6				04			Determine when plant conditions allow the termination of Standby Liquid Control with multiple control rods stuck out and what is the basis.	4.5		BOTH	EK1.04: 3.4 EK1.05: 3.4 EA2.03: 4.3	NEW	
500000 High Containment Hydrogen Conc. / V CFR41.10/43.5						2. 1. 20	Given applicable SOP, graph, and plant conditions, determine the power settings for the Hydrogen Recombiners and time to full power.	4.3		BOTH	EA1.03: 3.4	NEW	←Generics
295025 High Reactor Pressure / III CFR41.5/43.2			02				Describe the basis for a Reactor Recirculation Pump trip on High Reactor Pressure.	3.9		BOTH		NEW	
PAGE 2 TOTAL TIER 1 GROUP 1	0	0	1	1	0	1	PAGE TOTAL # QUESTIONS	3					
PAGE 1 TOTAL TIER 1 GROUP 1	3	3	3	0	1	0	PAGE TOTAL # QUESTIONS	10					
K/A CATEGORY TOTALS:	3	3	4	1	1	1	TIER 1 GROUP 1 GROUP POINT TOTAL	13					

←500000 random selection was EK1 which the only topic was Containment integrity, changed to generic 2.1.20 for execution of procedural steps concerning Hydrogen Recombiners.

E/APE#/NAME/SAFETY FUNCTION	K 1	K 2	K 3	A 1	A 2	G	TOPIC(S)	IMP	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
295001 Partial or Complete Loss of Forced Core Flow Circulation / I & IV CFR41.10/41.5/43.5						2. 4. 11	Given plant conditions and the power to flow map, determine the actions to be taken.	3.4		BOTH	2.4.1: 4.3 AA2.01: 3.5	NEW	↑ moved AA2 to generics
295002 Loss of Main Condenser Vacuum / III CFR41.4/43.2			05				Describe the basis for the isolation of the Main Steam Isolation Valves on a loss of condenser vacuum.	3.4		BOTH	AA1.04: 3.3	NEW	→ moved AA1 to AK3
295003 Partial or Complete Loss of AC Power/ VI CFR41.7					05		Given plant conditions and a Normal electrical line up and a loss of power determine the availability of safe shutdown systems and buses.	3.9		BOTH	AA1.03: 4.4	NEW	
295004 Partial or Total Loss of DC Power / VI CFR41.7					04		Describe the effects of a loss of DC power on the actuation of ECCS systems.	3.2		BOTH		NEW	
295008 High Reactor Water Level / II CFR41.4/41.5	01						Identify the affects of a High Reactor Water Level on the Main Turbine and Reactor Feed Pump Turbines.	3.0		BOTH		NEW	
295011 High Containment Temperature / V													
295012 High Drywell Temperature / V CFR41.9				02			Given plant conditions, determine the means of Drywell heat removal.	3.8		BOTH	AA1.01: 3.5	NEW	
295013 High Suppression Pool Temp. / V CFR41.9/41.10				02			Given plant conditions, determine the cause of rising Suppression Pool Temperature.	3.9		BOTH		NEW	
295016 Control Room Abandonment / VII CFR41.7		01					Given conditions and the attachment to the Shutdown from Remote Shutdown Panels ONEP, determine the condition of selected equipment.	4.4		BOTH		NEW	
295017 High Offsite Release Rate / IX CFR41.11		04					Given plant conditions, determine the monitoring of a plant release.	3.1		BOTH			
295018 Partial or Total Loss of CCW / VIII CFR41.4	01						Determine the effects on the RWCU with a sustained loss of Service Water.	3.5		BOTH		NEW	
PAGE 1 TOTAL TIER 1 GROUP 2	2	2	1	2	2	1	PAGE TOTAL # QUESTIONS	10					

↑ Moved from AA2 which was random selection to generics of AOP operations.
→ Moved from AA1 which was random selection to AK3 to determine basis for MSIV Isolation on loss of condenser vacuum AA1 topics tend to be lower level knowledge.

E/APE#/NAME/SAFETY FUNCTION	K 1	K 2	K 3	A 1	A 2	G	TOPIC(S)	IMP	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
295019 Partial or Total Loss of Inst. Air / VIII													
295020 Inadvertent Cont. Isolation / V & VII CFR41.9				01			Given an isolation of the Containment, determine the ability to reset and unisolate plant systems.	3.6		BOTH			
295022 Loss of CRD Pumps / I CFR41.5		04					Given plant conditions, describe the affects on reactor water level a loss of both Control Rod Drive Pumps.	2.5		BOTH		NEW	
295026 Suppression Pool High Water Temp. / V CFR41.7/41.9/41.10/41.14/43.5		01					Given plant conditions, determine when Suppression Pool cooling is no longer effective and alternate actions are required.	3.9		BOTH		NEW	
295027 High Containment Temperature / V CFR41.5/41.9/41.10/43.5		03					Given plant conditions, identify the cooling mechanisms for the Containment.	3.5		BOTH	EK2.01: 3.2	NEW	
295028 High Drywell Temperature / V CFR41.5/41.10/41.14/43.5					03		Given plant conditions, determine the availability of reactor water level indications.	3.7		BOTH	2.4.20: 3.3	NEW	↓ moved EA1 to EA2
295029 High Suppression Pool Water Level / V CFR41.7			01				Describe the basis for emergency depressurizaion of the reactor in a High Suppression Pool level condition.	3.5		BOTH	2.4.18: 2.7	NEW	
295030 Low Suppression Pool Water Level / V CFR41.9						2. 4. 18	Describe the basis for the low level limit for emergency depressurization based on Suppression Pool Level.	2.7		BOTH	EK3.01: 3.8	NEW	° moved from A1 to Generics
295033 High Secondary Containment Area Radiation Levels / IX CFR41.11/41.12/43.4			03				Given plant conditions, determine the plant response to conditions to control offsite radiation levels	3.8		BOTH		NEW	
295034 Secondary Containment Ventilation High Radiation / IX													
295038 High Offsite Release Rate / IX CFR41.7/43.4			03				Determine the conditions which would result in an isolation of the Control Room Ventilation System (with regard to High Radiation).	3.7		BOTH			
600000 Plant Fire On Site / VIII													
PAGE 2 TOTAL TIER 1 GROUP 2	0	3	3	1	1	1	PAGE TOTAL # QUESTIONS	9					
PAGE 1 TOTAL TIER 1 GROUP 2	2	2	1	2	2	1	PAGE TOTAL # QUESTIONS	10					
K/A CATEGORY TOTALS:	2	5	4	3	3	2	TIER 1 GROUP 2 GROUP POINT TOTAL	19					

↓ random selection for 295012 and 295028 were both A1, 295028 was moved to A2 to allow different knowledge about High Drywell Temperature to be tested.
° random selection for 295030 was A1, this was moved to generics based on number of questions in A1 and allowance for basis question on Suppression pool low level.

E/APE#/NAME/SAFETY FUNCTION	K 1	K 2	K 3	A 1	A 2	G	TOPIC(S)	IMP	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
295021 Loss of Shutdown Cooling / IV CFR41.5/41.7/43.5				04			Determine an approved method of Alternate decay heat removal.	3.7		BOTH		NEW	Mode 3
295023 Refueling Accidents / VIII CFR41.2/41.10/41.12/43.4/43.5/43.6/43.7			01				Describe personnel response to fuel handling accident.	3.6		BOTH		NEW	
295032 High Secondary Containment Area Temperature / V CFR41.4/41.9/41.10/43.5				05			Given plant conditions, determine systems which should have isolated.	3.7		BOTH		NEW	
295035 Secondary Containment High Differential Pressure / V CFR41.7/41.13/43.4			02				Describe the operation of the Secondary Containment Ventilation system and its response to an abnormal differential pressure in Secondary Containment.	3.3		BOTH			
295036 Secondary Containment High Sump/Area Water Level / V													
K/A CATEGORY TOTALS:	0	0	2	2	0	0	TIER 1 GROUP 3 GROUP POINT TOTAL	4					

SYSTEM#/NAME	K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G	TOPIC(S)	IMP	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
201001 CRD Hydraulic CFR41.6							03					Describe the reaction of the CRD System flows when moving a control rod.	2.9		BOTH		NEW	± moved from K5 to A1
201005 RCIS CFR41.6								13				Given an alarm and RCIS indications, determine cause based on system lights.	3.8		BOTH		NEW	Group 1 or 2 rod drift
202002 Recirculation Flow Control CFR41.6									01			Given plant conditions, identify the plant response to a Recirc Flow Control Runback.	3.6		BOTH		NEW	Recirc Pumps in abnormal pump configuration
203000 RHR/LPCI: Injection Mode CFR41.7						06						Describe the response of the RHR (LPCI) system with a low Suppression Pool Water Level.	3.8		BOTH	K1.02: 3.9	NEW	
209001 LPCS CFR41.7/41.8										01		Describe the operation of the LPCS pump if the system is manually overridden with an initiation signal.	3.8		BOTH	A4.03: 3.7 A4.05: 3.8	NEW	
209002 HPCS CFR41.7/41.8		03										Describe the operation of HPCS following an initiation with a loss of AC power.	2.8		BOTH		MOD	NRC 98 – 19 conditions changed
211000 SLC CFR41.6/41.7				02								Describe the response of the SLC system when aligned for test tank operation and a SLC initiation signal.	3.0		BOTH	A2.06: 3.1 A2.07: 2.9	NEW	CR at GGNS
212000 RPS CFR41.6											2. 1. 31	Given plant conditions and indications, determine the probable cause (Blown Fuse causes partial reactor scram).	4.2		BOTH	2.4.48: 3.5 A2.19: 3.8	NEW	Blown fuse indication ″ moved from A3 to generics
215003 IRM CFR41.6			01									Given IRM conditions, determine the status of control rod blocks and RPS.	3.9		BOTH	K3.04: 3.6 A1.03: 3.6 A1.04: 3.4		
215004 Source Range Monitor CFR41.6								02				Determine the ability to continue a reactor startup with more than one SRM in a Division failed.	3.4		BOTH	A2.03: 3.0	NEW	
PAGE 1 TOTAL TIER 2 GROUP 1	0	1	1	1	0	1	1	2	1	1	1	PAGE TOTAL # QUESTIONS	10					

± 201001 random selection K5 moved to A1 due to low importance factors of K5 topics and number of K5s selected on random.
″ 212000 random selection A3 moved to generics to encompass indication of failure which can cause an inadvertent partial reactor scram.

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
215005 APRM / LPRM CFR41.6								04				Determine RPS actuation and rod blocks based on LPRM inputs.	3.8		BOTH	A1.02: 3.9 A1.03: 3.6 A1.04: 4.1 A2.03: 3.6	MOD	NRC 98-15
216000 Nuclear Boiler Instrumentation CFR41.7				05								Describe logic required to initiate ECCS.	3.9		BOTH		NEW	
217000 RCIC CFR41.7/41.10					06							Given RCIC indications following a RCIC Turbine shutdown/trip, determine the cause of the RCIC failure.	2.7		BOTH	A4.02: 3.9 A2.02: 3.8	NEW	
218000 ADS CFR41.7/41.8					01							Given plant conditions, determine the operation of ADS.	3.8		BOTH	K4.02: 3.8 K4.03: 3.8	NEW	
223001 Primary CTMT and Auxiliaries CFR41.7										07		Given plant conditions, determine the operation of the Drywell vacuum breaker valves. (Drywell to Containment pressure differential)	4.2		BOTH	A4.06: 4.0 A3.02: 3.4	NEW	
223002 PCIS / Nuclear Steam Supply Shutoff CFR41.7/41.9/43.5							01					Given isolation conditions, evaluate indications to determine whether appropriate actions have occurred.	3.5		BOTH	A3.01: 3.4 2.4.46: 3.5 2.4.48: 3.5	NEW	
239002 SRVs CFR41.3		01										Identify the power supply to the SRVs (ADS, Low-Low Set, and Normal).	2.8		BOTH		NEW	
241000 Reactor / Turbine Pressure Regulator CFR41.5						11						Describe the response of the plant with a failure of the Main Stop and Control Valves closed with the reactor at power.	3.4		BOTH	A1.01: 3.9 A1.02: 4.1 A1.07: 3.8	NEW	
259001 Reactor Feedwater CFR41.4			01									Given plant startup/heatup with Feedwater aligned to feed the reactor and a subsequent trip of the Reactor Feed pump, determine the action of Reactor Water Level.	3.9		BOTH		NEW	
259002 Reactor Water Level Control CFR41.5				10								Describe the operation of the Feed water Level Control System on a failure of the Feed Flow input signal.	3.4		BOTH	K6.04: 3.1	NEW	
PAGE 2 TOTALS TIER 2 GROUP 1	0	1	1	2	2	1	1	1	0	1	0	PAGE 2 TOTAL # QUESTIONS	10					

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
261000 SGTS CFR41.17/41.11	08											Describe the response of the radiation monitoring system for SBGT on an initiation of the system.	2.8		BOTH	K4.01: 3.7	NEW	
264000 EDGs CFR41.8									01			Given plant conditions determine the type of operations expected for the Division I, II and HPCS Diesel Generators.	3.0		BOTH		MOD	
264000 EDGs CFR41.8			01									Describe the reaction of the Reactor Core Isolation Cooling (RCIC) System with a failure of the Division I Diesel Generator to start on a loss of 15AA.	4.2		**RO**		NEW	
203000 RHR/LPCI: Injection Mode CFR41.10									01			With the plant aligned for shutdown cooling and a LOCA, describe the alignment of RHR for LPCI operation.	3.8		BOTH	A3.08: 4.1 A4.02: 4.1 K4.01: 4.2		Manual realignment requirement.
209002 HPCS CFR41.7										02		Describe the system responses to manual realignment of HPCS suction from the Suppression Pool to the CST following an automatic swap and subsequent operations.	3.6		**RO**	A2.12: 3.3 A2.13: 3.4		
201005 RCIS CFR41.2/41.6											2. 2. 25	Describe the limitations on control rod movement with the reactor at full power and the bases for these limitations.	2.5		**RO**	2.2.2: 4.0 K4.03: 3.5 K4.05: 3.5		× Reactivity Management
218000 ADS CFR41.7								03				With plant conditions requiring operation of ADS and a loss of Instrument Air, describe a method to open ADS valves.	3.4		**RO**	A1.03: 3.2 K4.04: 3.5 K6.04: 3.6		≥ moved from A4 to A2
259001 Reactor Feedwater CFR41.4						07						Describe the affects on the Reactor Feed Pumps on a loss of signal from theReactor Water Level Control System.	3.8		**RO**	259002 K3.02: 3.7		
PAGE 3 TOTALS TIER 2 GROUP 1	1	0	1	0	0	1	0	1	2	1	1	PAGE TOTAL # QUESTIONS	8					
PAGE 1 TOTALS TIER 2 GROUP 1	0	1	1	1	0	1	1	2	1	1	1	PAGE TOTAL # QUESTIONS	10					
PAGE 2 TOTALS TIER 2 GROUP 1	0	1	1	2	2	1	1	1	0	1	0	PAGE TOTAL # QUESTIONS	10					
K/A CATEGORY TOTALS:	1	2	3	3	2	3	2	4	3	3	2	TIER 2 GROUP 1 GROUP POINT TOTAL	28					

≥ Random selection was A4 changed to A2, previous question on ADS sufficiently covered topic A4.

× Random selection 223002 A3 was changed to 201005 Generics, due to A1 and A3 similarities of topic and A1 was previously used and a topic for reactivity management was needed.

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
201003 Control Rod and Drive Mechanism CFR41.2				01								Describe the feature of the Control Rod Drive Mechanism which limits the transient of a control rod drop accident and when would this transient be at a maximum effect.	2.9		**RO**		NEW	∝ moved from K6 to K4
202001 Recirculation CFR41.5/41.6/43.6					05							Given plant conditions, describe the operational sequence of an End of Cycle Recirculation Pump Trip.	3.5		BOTH		NEW	
204000 RWCU CFR41.6						07						Given plant conditions, determine operation of the Reactor Water Cleanup System upon an initiation of SLC B.	3.3		BOTH		NEW	
205000 Shutdown Cooling CFR41.7		02										Describe the affects of a loss of power to the RHR Shutdown Cooling suction valve with respect to interlocks with other RHR valves.	2.5		BOTH	A2.10: 2.9 A3.01: 3.2 K6.01: 3.3	NEW	
219000 RHR /LPCI Suppression Pool Cooling Mode CFR41.9			01									Describe the limitations concerning the RHR Heat Exchanger bypass and outlet valves.	3.9		**RO**	2.1.32: 3.4	NEW	
226001 RHR/LPCI: CTMT Spray Mode CFR41.9							01					Identify the response of Containment Pressure, Suppression Pool Temperature and level when Containment Spray is operated.	3.6		BOTH	A1.04: 3.3 A1.08: 3.1	NEW	①① moved from K5 to A1
239001 Main and Reheat Steam CFR41.4									01			Given plant conditions, determine the reaction of the Main and Reheat Steam system. (Isolation action)	4.2		BOTH		NEW	
245000 Main Turbine Gen. and Auxiliaries CFR41.4						05						Given plant conditions, determine a condition that would result in a Turbine Generator trip.	2.9		BOTH	K6.04: 2.6	NEW	Generator Hydrogen Syst. Mod
PAGE 1 TOTAL TIER 2 GROUP 2	0	1	1	1	1	2	1	0	1	0	0	PAGE TOTAL # QUESTIONS	8					

∝ Random selection was K6 which was covered on NRC exam 1998, changed to K4.
①① Random selection was K5, moved to A1 due to low discriminatory value of topic 226001 K5.

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
256000 Reactor Condensate CFR41.4									07			Describe the effects of a Low Pressure Feedwater Heater isolation on the Condensate System operation and plant operation.	2.9		**RO**	A3.04: 3.0 A3.01: 2.7 A2.08: 3.1	NEW	
262001 AC Electrical Distribution CFR41.4								03				Determine the effects of a ground fault on BOP Transformer 12A.	3.9		BOTH		NEW	
262002 UPS (AC/DC) CFR41.4	01											Determine the effect of a loss of UPS power on Reactor Water Level Control and Feedwater Systems.	2.8		**RO**	K1.02: 2.8	NEW	INFI 90 Control System
263000 DC Electrical Distribution CFR41.4			02									Discern the effects of a loss of DC power on the operation of circuit breakers.	3.5		BOTH		BANK	①②moved from K4 to K3
271000 Offgas CFR41.7/41.13				04								Discuss the effects an emergency shutdown of the Hydrogen Water Chemistry System on the Offgas system.	3.3		BOTH	A2.07: 2.7 A1.14: 2.7 K6.06: 2.5	NEW	①③ moved from K3 to K4
272000 Radiation Monitoring CFR41.11/41.13/43.4					01							Discuss the effects of injection of Hydrogen into the Condensate system with regard to plant radiation levels.	3.2		**RO**		NEW	Hydrogen Water Chemistry
286000 Fire Protection CFR41.4									04			Given conditions requiring the initiation of fire deluge systems for a Containment Filter Train, describe how this in accomplished.	3.2		BOTH	2.1.30: 3.9 2.4.25: 2.9 2.4.27: 3.0	NEW	Manual system
290001 Secondary CTMT CFR41.9						09						Describe the ability of Auxiliary Building Fire Protection system to be restored following an Auxiliary Building Isolation in conjunction with a loss of AC Power.	3.4		BOTH	A2.06: 3.7 286000 A2.09: 2.7	NEW	
PAGE 2 TOTAL TIER 2 GROUP 2	1	0	1	1	1	1	0	1	2	0	0	PAGE TOTAL # QUESTIONS	8					

①② Random selection was K4 that is N/A at GGNS, changed to K3.02.
①③Random selection was K3, changed to K4.04 to allow question concerning precautions about newly installed Hydrogen Water Chemistry System and hazards.

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
290003 Control Room HVAC CFR41.4										01		Given plant conditions, determine the appropriate Control Room HVAC configuration.	3.2		BOTH	A3.01: 3.3		
300000 Instrument Air CFR41.4			02									Given a loss of Instrument Air, determine the affect on the ability to feed water to the reactor.	3.3		BOTH		NEW	①④Random selection replaced with K3.02.
400000 Component Cooling Water CFR41.4							02					Given plant conditions, determine the affects on CCW and alternate actions to be taken. (Loss of Plant Service Water No LOCA.)	2.8		BOTH	K1.01: 3.2	NEW	GGNS Event 10/99
PAGE 3 TOTALS	0	0	1	0	0	0	1	0	0	1	0	PAGE 3 TOTAL # QUESTIONS	3					
PAGE 1 TOTALS	0	1	1	1	1	2	1	0	1	0	0	PAGE 1 TOTAL # QUESTIONS	8					
PAGE 2 TOTALS	1	0	1	1	1	1	0	1	2	0	0	PAGE 2 TOTAL # QUESTIONS	8					
K/A CATEGORY TOTALS:	1	1	3	2	2	3	2	1	3	1	0	TIER 2 GROUP 2 GROUP POINT TOTAL	19					

①④ Random selection was K2. K2 only had two topics which had no discretionary value. Replaced with K3.02 dealing with system interrelations.

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
215001 Traversing In-core Probe																		
233000 Fuel Pool Cooling and Cleanup CFR41.4/41.10					01							Given plant conditions post refueling, determine the method of heat removal from the Fuel Pool Cooling Heat Exchangers.	2.5		BOTH		NEW	
234000 Fuel Handling Equipment CFR41.4/43.7	06											Determine the ability to operate the Refueling Bridge given plant conditions. (RCIS interlocks)	3.0		**RO**		NEW	
238003 MSIV Leakage Control																		
268000 Radwaste CFR41.13/43.4											2. 4. 21	Given conditions and a Liquid Radwaste Discharge in Progress, determine whether a release will continue.	3.7		BOTH		MOD	①⑤ moved from K3 to generics
288000 Plant Ventilation																		
290002 Reactor Vessel Internals CFR41.3/41.14/43.2					05							Given plant conditions, determine the allowances based on the Pressure vs Temperature graph for the Reactor Vessel.	3.1		BOTH	2.1.25: 2.8	NEW	①⑥ moved from K1 to K5
K/A CATEGORY TOTALS:	1	0	0	0	2	0	0	0	0	0	1	TIER 2 GROUP 3 GROUP POINT TOTAL	4					

①⑤ Random selection was K3, Topic K3 does not have any discriminatory value, changed to generics 2.4.21.
①⑥ Random selection K1 topics sufficiently covered in other areas of examination, moved to K5 to evaluate Brittle Fracture Limits

CATEGORY	C1	C2	C3	C4	TOPIC(S)	IMP	REC #	SRO/RO /BOTH	RELATED K/A	ORIGIN	NOTES:
CONDUCT OF OPERATIONS – Working Hour Limits CFR41.10	2.1.2				Determine the maximum number of working hours and allowances per conduct of operations.	3.0		**RO**		NEW	
CONDUCT OF OPERATIONS – Locked Compnts CFR41.10	2.1.1				Evaluate components to determine requirements to be locked.	3.7		**RO**		NEW	
CONDUCT OF OPERATIONS – Procedure Usage CFR41.10	2.1.23				Evaluate the allowances for concurrent performance of sections of the Integrated Operating Instructions.	3.9		**RO**	2.1.20: 4.3 2.2.1: 3.7	NEW	
CONDUCT OF OPERATIONS – Tech Specs CFR41.10/43.2	2.1.12				Given Technical Specifications, apply actions for a given situation.	2.9		**RO**		NEW	
EQUIPMENT CONTROL – Procedures CFR41.10		2.2.11			Determine the requirements for independent verification and requirements of Temporary Alteration installation.	2.5		**RO**		NEW	
EQUIPMENT CONTROL – Safety Limits CFR41.10/43.2/43.6		2.2.22			Given plant conditions, evaluate the status of plant Safety Limits.	3.4		**RO**	2.1.10: 2.7 2.1.11: 3.0 2.1.33: 3.4	MOD	Apply new definition of Adequate Core Cooling
RADIATION CONTROL - Radiation Limits CFR41.10/43.4			2.3.4		Given personnel radiation exposure, determine their availability based on GGNS Radiation limits.	2.5		**RO**		NEW	
RADIATION CONTROL – ALARA CFR41.10/43.4			2.3.2		Describe the personnel hazard of operating RHR Suppression Pool Cooling mode. (ALARA)	2.5		**RO**	2.1.32: 3.4	NEW	Internal Hazard to personnel
EMERGENCY PROCEDURES / PLAN – TBCW CFR41.10/43.5				2.4.24	Determine actions to be taken for given conditions with regard to a loss of TBCW.	3.3		**RO**	2.4.11: 3.4 2.4.49: 4.0	NEW	Loss of TBCW ONEP
EMERGENCY PROCEDURES / PLAN – EOPs CFR41.10/43.5				2.4.4	Given plant conditions, identify the Emergency Operating Procedures requiring implementation.	4.0		**RO**		NEW	Entry into SAP
EMERGENCY PROCEDURES / PLAN – EOPs CFR41.10/43.5				2.4.20	Given plant conditions, determine the actions to be taken with regard to the Emergency Operating Procedures.	3.3		**RO**		NEW	
EMERGENCY PROCEDURES / PLAN – Diagnostics CFR41.5				2.4.21	Given plant parameters and elevated in-leakage into the Drywell, determine the status of reactor coolant system integrity.	3.7		**RO**		NEW	GGNS Drywell air leak
EMERGENCY PROCEDURES / PLAN - Fire CFR41.10/43.13/43.4/43.5				2.4.25	Given a fire in the Offgas Charcoal Beds, identify the actions to be taken.	2.9		**RO**		NEW	Has occurred at GGNS and possible occurrence
K/A CATEGORY TOTALS:	4	2	2	5	TIER 3 GROUP POINT TOTAL	13					

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

Examiners: _____ Operators: _____
_____ _____
_____ _____

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

1. Shift Recirculation Pumps to fast speed.
2. Raise Recirculation flow to 60 % core flow.
3. Raise Reactor Power by withdrawing control rods.
4. Perform operator actions for a stuck control rod per ONEP.
5. Analyze the impact of a failure of the reference leg of instruments connected to D004A and apply Technical Specifications.
6. Initiate a reactor scram based on rising Drywell Pressure.
7. Respond to a failure of Division 1 ECCS failure to initiate.
8. With a failure of Feedwater Line in the Turbine Building and reduced injection systems maintain reactor level per the EOPs.

Initial Conditions: Reactor Power is at 34 % bringing the plant up following an outage; Reactor Recirculation pumps in Slow Speed; a single Reactor Feed Pump in single element Master Level Control;

INOPERABLE Equipment

APRM 'F' is INOP due to a failed power supply card
ESF 12 Transformer is tagged out of service for maintenance
CCW Pump 'C' is tagged out of service for pump seal replacement
RHR 'C' Pump is tagged out of service for motor oil replacement

Appropriate clearances and LCOs are written.

Turnover: Continue to bring the plant to full power per IOI-2. There is severe weather reported in the Tensas Parish area.

Event No.	Malf. No.	Event Type*	Event Description
1		N, R (RO)	Shift Reactor Recirculation Pumps to Fast Speed (SOI 04-1-01-B33-1)
2		R (RO)	Withdraw control rods to increase power. (Control Rod Pull Sheet)
3		C (RO, BOP)	Control Rod XX-YY is stuck, un-stick control rod per ONEP. (ONEP 05-1-02-IV-1)
4		I (RO, BOP)	Respond to a failure of the reference leg of D004A level instrument. (Tech Specs)

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

Event No.	Malf. No.	Event Type*	Event Description
5			Initiate manual reactor scram in response to rising Drywell pressure. (05-1-02-I-1)
6		I (BOP)	Failure of Division 1 ECCS to automatically initiate on High Drywell Pressure
7		M (ALL)	Feedwater Line rupture in the Turbine Building with leakage from the reactor
8		C (BOP)	HPCS pump trip on initiation
9		C (BOP)	RCIC fails to achieve rated conditions for injection

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

Critical Tasks

- Lower reactor pressure to allow injection from low pressure systems
- Recognize failure of Division 1 to initiate and manually initiate Division 1

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description:

Time	Position	Applicant's Actions or Behavior

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

Examiners: _____ Operators: _____
_____ _____
_____ _____

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

1. Shift Recirculation Pumps to fast speed.
2. Raise Recirculation flow to 60 % core flow.
3. Raise Reactor Power by withdrawing control rods.
4. Perform operator actions for a stuck control rod per ONEP.
5. Analyze the impact of a failure of the reference leg of instruments connected to D004A and apply Technical Specifications.
6. Initiate a reactor scram based on rising Drywell Pressure.
7. Respond to a failure of Division 1 ECCS failure to initiate.
8. With a failure of Feedwater Line in the Turbine Building and reduced injection systems maintain reactor level per the EOPs.

Initial Conditions: Reactor Power is at 34 % bringing the plant up following an outage; Reactor Recirculation pumps in Slow Speed; a single Reactor Feed Pump in single element Master Level Control;

INOPERABLE Equipment

APRM 'F' is INOP due to a failed power supply card
ESF 12 Transformer is tagged out of service for maintenance
CCW Pump 'C' is tagged out of service for pump seal replacement
RHR 'C' Pump is tagged out of service for motor oil replacement

Appropriate clearances and LCOs are written.

Turnover: Continue to bring the plant to full power per IOI-2. There is severe weather reported in the Tensas Parish area.

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

Event No.	Malf. No.	Event Type*	10CFR55.45a	K/A	Event Description
1		N, R (RO)	1, 2, 4, 5, 7	202001 A4.01; A4.02; A4.04 202002 A4.07; A4.08	Shift Reactor Recirculation Pumps to Fast Speed (SOI 04-1-01-B33-1)
2		R (RO)	1, 2, 5, 6	201005 A3.01; A3.02; A3.03; A4.01 2.2.2	Withdraw control rods to increase power. (Control Rod Pull Sheet)
3		C (RO, BOP)	3, 5, 6	201005 A3.03 201001 A4.04 2.4.4; 2.4.11; 2.4.48	Control Rod XX-YY is stuck, un-stick control rod per ONEP. (ONEP 05-1-02-IV-1)
4		I (RO, BOP)	3, 4	2.1.12; 2.4.45; 2.4.48 295010 AA2.02; AA2.06	Respond to a failure of the reference leg of D004A level instrument. (Tech Specs)
5			5, 6, 12	2.4.4; 2.4.49 295006 AA1.01; AA1.05; AA1.07	Initiate manual reactor scram in response to rising Drywell pressure. (05-1-02-I-1)
6		I (BOP)	3, 4, 7, 10	2.4.4 295024 EA1.0	Failure of Division 1 ECCS to automatically initiate on High Drywell Pressure
7		M (ALL)	7, 13	295031 EA1.0 203000 A3.08 241000 A4.06	Feedwater Line rupture in the Turbine Building with leakage from the reactor
8		C (BOP)	3	295031 EA1.04 209002 A2.02	HPCS pump trip on initiation
9		C (BOP)	3	295031 EA1.05 217000 A2.10; A3.02; A4.01	RCIC fails to achieve rated conditions for injection

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

Critical Tasks

- Lower reactor pressure to allow injection from low pressure systems

- Recognize failure of Division 1 to initiate and manually initiate Division 1

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description:

Time	Position	Applicant's Actions or Behavior

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

Examiners: _____ Operators: _____
_____ _____
_____ _____

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

1. Raise Reactor Power by withdrawing control rods.
2. Place 2 nd Reactor Feed Pump on Master Level Control.
3. Respond to a failure of APRM 'B' downscale and apply appropriate Technical Specifications.
4. Respond to the report of a leak on the Hydraulic Power Unit for Reactor Feed Pump 'B'
5. Respond to the above core plate pressure instrument line rupture which causes Drywell pressure to rise.
6. Respond to ATWS with a failure of the Main Steam Bypass valves to open per EOPs.
7. Respond to a failure of Standby Liquid Control to function.

Initial Conditions: Reactor Power is at 44 % bringing the plant up following an outage; Reactor Recirculation pumps in Fast Speed; a single Reactor Feed Pump in three element Master Level Control;

INOPERABLE Equipment

APRM 'F' is INOP due to a failed power supply card
ESF 12 Transformer is tagged out of service for maintenance
CCW Pump 'C' is tagged out of service for pump seal replacement
RHR 'C' Pump is tagged out of service for motor oil replacement

Appropriate clearances and LCOs are written.

Turnover: Continue to bring the plant to full power per IOI-2. There is severe weather reported in the Tensas Parish area.

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

Event No.	Malf. No.	Event Type*	10CFR55.45a	K/A	Event Description
1		R (RO)	1, 2, 4, 5	201005 A3.01; A3.02; A3.03; A4.01 2.2.2	Raise Reactor power using control rods to between 45 and 55% (Control Rod Pull Sheet)
2		N (RO)	2, 4, 5, 6	259001 A4.02 259002 A3.09; A4.02; A4.03	Place Reactor Feed Pump 'B' in service on Master Level Control. (SOI 04-1-01-N21-1)
3		I (RO)	3, 5	215005 A2.02 2.1.12; 2.1.33	Respond to a failure of APRM 'B' downscale and apply appropriate Technical Specifications.
4		C (RO)	3, 4, 7	295009 AA1.01 2.4.4;	Respond to a leak on the Hydraulic Power Unit For RFPT 'B'.
5		I (RO, BOP)	3, 4, 5, 6, 13	2.4.46; 2.4.47; 2.4.48 295010 AA2.02; AA2.06	Respond to failure of the Above Core Plate Pressure Detector instrument line.
6			2, 3, 4, 7	2.4.4; 2.4.49 295006 AA1.01; AA1.05; AA1.07	Initiate manual reactor scram in response to rising Drywell pressure. (05-1-02-I-1)
7		M (ALL)	6, 8, 12, 13	295037 EA1.0; EA2.0 203000 A3.08 241000 A4.06	ATWS
8		C (RO)	3, 4	295007 AA1.04 295025 EA1.03 239001 A2.06 241000A2.03	Respond to failure of the Main Steam Bypass Valves to open.
9		C (BOP)	3, 4	295037 EA1.04; EA1.10 211000 A1.0; A2.04; A3.0	Respond to failure of Standby Liquid Control to inject into the reactor.

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

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

Critical Tasks

- Insert Control Rods in response to ATWS conditions.
- Inject Standby Liquid Control prior to Suppression Pool Temperature reaches 110 °F.
- Identify the need for Alternate Standby Liquid Control injection.
- 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 when reactor level reaches –192”.

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description:

Time	Position	Applicant's Actions or Behavior

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

Examiners: _____ Operators: _____
_____ _____
_____ _____

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

1. Raise Reactor Power by withdrawing control rods.
2. Place 2 nd Reactor Feed Pump on Master Level Control.
3. Respond to a failure of APRM 'B' downscale and apply appropriate Technical Specifications.
4. Respond to the report of a leak on the Hydraulic Power Unit for Reactor Feed Pump 'B'
5. Respond to the above core plate pressure instrument line rupture which causes Drywell pressure to rise.
6. Respond to ATWS with a failure of the Main Steam Bypass valves to open per EOPs.
7. Respond to a failure of Standby Liquid Control to function.

Initial Conditions: Reactor Power is at 44 % bringing the plant up following an outage; Reactor Recirculation pumps in Fast Speed; a single Reactor Feed Pump in three element Master Level Control;

INOPERABLE Equipment

APRM 'F' is INOP due to a failed power supply card
ESF 12 Transformer is tagged out of service for maintenance
CCW Pump 'C' is tagged out of service for pump seal replacement
RHR 'C' Pump is tagged out of service for motor oil replacement

Appropriate clearances and LCOs are written.

Turnover: Continue to bring the plant to full power per IOI-2. There is severe weather reported in the Tensas Parish area.

Event No.	Malf. No.	Event Type*	Event Description
1		R (RO)	Raise Reactor power using control rods to between 45 and 55% (Control Rod Pull Sheet)
2		N (RO)	Place Reactor Feed Pump 'B' in service on Master Level Control. (SOI 04-1-01-N21-1)
3		I (RO)	Respond to a failure of APRM 'B' downscale and apply appropriate Technical Specifications.

4		C (RO)	Respond to a leak on the Hydraulic Power Unit For RFPT ‘B’.

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

Event No.	Malf. No.	Event Type*	Event Description
5		I (RO, BOP)	Respond to failure of the Above Core Plate Pressure Detector instrument line.
6			Initiate manual reactor scram in response to rising Drywell pressure. (05-1-02-I-1)
7		M (ALL)	ATWS
8		C (RO)	Respond to failure of the Main Steam Bypass Valves to open.
9		C (BOP)	Respond to failure of Standby Liquid Control to inject into the reactor.

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

Critical Tasks

- Insert Control Rods in response to ATWS conditions.
- Inject Standby Liquid Control prior to Suppression Pool Temperature reaches 110 °F.
- Identify the need for Alternate Standby Liquid Control injection.
- 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 when reactor level reaches –192”.

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description:

Time	Position	Applicant's Actions or Behavior

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **3** Op-Test No.: **Day 2**

Examiners: _____ Operators: _____
_____ _____
_____ _____

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

1. Lower Reactor Power using Reactor Recirculation Flow Control.
2. Place Standby Service Water 'B' in operation for a chemical addition.
3. Respond to a failure of the Main Generator Voltage Regulator.
4. Respond to a trip of the Standby Service Water Pump 'B' and apply Technical Specifications.
5. Respond to a steam leak in the Auxiliary Building Steam Tunnel which will not completely isolate.
6. Respond to ATWS (no challenge to Containment).
7. Respond to plant parameters per EOP-4 Secondary Containment Control.
8. Respond to lowering reactor level with limited high pressure injection.

Initial Conditions: Reactor Power is at 100 %

INOPERABLE Equipment

APRM 'F' is INOP due to a failed power supply card
ESF 12 Transformer is tagged out of service for maintenance
CCW Pump 'C' is tagged out of service for pump seal replacement
RHR 'C' Pump is tagged out of service for motor oil replacement

Appropriate clearances and LCOs are written.

Turnover: Lower Reactor power to attain less than 1173 MWe and less than 40 % valve position on the lead Turbine Control Valve in preparation for performing the Turbine Stop and Control Valve Operability Surveillance. There is severe weather reported in the Tensas Parish area.

Scenario **3** Day **2** (Continued)

Event No.	Malf. No.	Event Type*	10CFR55.45a	K/A	Event Description
1		R (RO)	2, 4, 5, 6	202001 A4.04 202002 A4.08 2.2.2	Lower Reactor power using Recirculation Flow Control to < 1173 MWe
2		N (BOP)	4, 5, 6	2.1.30	Place Standby Service Water Pump 'B' in operation for Chemical Addition. (SOI 04-1-01-P41-1)
3		I (RO)	3, 4, 5	245000 A3.10; A4.12; A4.14 2.1.30; 2.4.10	Respond to a failure of the Main Generator Voltage Regulator.
4		C (BOP)	3, 4	2.4.48; 2.1.12	Respond to a trip of Standby Service Water Pump 'B' and apply Technical Specifications.
5		M (ALL)	3, 4, 6, 13	2.4.46; 2.4.47; 2.4.48; 2.4.49 290001 A2.06; A4.04 295032 EA1.02; EA1.05	Respond to steam leak in the Auxiliary Building Steam tunnel (unisolable – leakage past MSIVs will be indicated).
6			2, 3, 4, 8	2.4.4; 2.4.49 295006 AA1.01; AA1.05; AA1.07	Initiate manual reactor scram due to Steam Leak in the Auxiliary Building. (05-1-02-I-1)
7		C (RO)	4, 6, 12, 13	295037 EA1.0; EA2.0	ATWS (three control rods stuck full out)
8		C (BOP)	3, 4, 7	295031 EA1.0; EA2.0 259001 A2.04	Respond to loss of Feedwater with a failure HPCS (HPCS Pump will trip on start attempt).

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

Critical Tasks

- Isolate the main steam lines
- Manually scram the reactor
- Maintain adequate core cooling (loss of normal feed with degraded ECCS capability)

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description:

Time	Position	Applicant's Actions or Behavior

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **3** Op-Test No.: **Day 2**

Examiners: _____ Operators: _____
_____ _____
_____ _____

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

1. Lower Reactor Power using Reactor Recirculation Flow Control.
2. Place Standby Service Water 'B' in operation for a chemical addition.
3. Respond to a failure of the Main Generator Voltage Regulator.
4. Respond to a trip of the Standby Service Water Pump 'B' and apply Technical Specifications.
5. Respond to a steam leak in the Auxiliary Building Steam Tunnel which will not completely isolate.
6. Respond to ATWS (no challenge to Containment).
7. Respond to plant parameters per EOP-4 Secondary Containment Control.
8. Respond to lowering reactor level with limited high pressure injection.

Initial Conditions: Reactor Power is at 100 %
INOPERABLE Equipment
APRM 'F' is INOP due to a failed power supply card
ESF 12 Transformer is tagged out of service for maintenance
CCW Pump 'C' is tagged out of service for pump seal replacement
RHR 'C' Pump is tagged out of service for motor oil replacement
Appropriate clearances and LCOs are written.

Turnover: Lower Reactor power to attain less than 1173 MWe and less than 40 % valve position on the lead Turbine Control Valve in preparation for performing the Turbine Stop and Control Valve Operability Surveillance. There is severe weather reported in the Tensas Parish area.

Event No.	Malf. No.	Event Type*	Event Description
1		R (RO)	Lower Reactor power using Recirculation Flow Control to < 1173 MWe
2		N (BOP)	Place Standby Service Water Pump 'B' in operation for Chemical Addition. (SOI 04-1-01-P41-1)
3		I (RO)	Respond to a failure of the Main Generator Voltage Regulator.

4		C (BOP)	Respond to a trip of Standby Service Water Pump ‘B’ and apply Technical Specifications.

Scenario **3** Day **2** (Continued)

Event No.	Malf. No.	Event Type*	Event Description
5		M (ALL)	Respond to steam leak in the Auxiliary Building Steam tunnel (unisolable – leakage past MSIVs will be indicated).
6			Initiate manual reactor scram due to Steam Leak in the Auxiliary Building. (05-1-02-I-1)
7		C (RO)	ATWS (three control rods stuck full out)
8		C (BOP)	Respond to loss of Feedwater with a failure HPCS (HPCS Pump will trip on start attempt).

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

Critical Tasks

- Isolate the main steam lines
- Manually scram the reactor
- Maintain adequate core cooling (loss of normal feed with degraded ECCS capability)

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description:

Time	Position	Applicant's Actions or Behavior

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **4** Op-Test No.: **BACK UP 1**

Examiners: _____ Operators: _____
_____ _____
_____ _____

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

1. Raise power 5% with Recirc flow.
2. Startup one loop of Suppression Pool Cooling.
3. Implement Technical Specifications for LPCI inoperability of an RHR loop When it is running in suppression pool cooling.
4. Implement the AOP(ONEP) for a Recirc Pump trip.
5. Respond to a Seal Steam Controller failure.
6. Execute the EOPs for a Feedwater line break in the Drywell with check Valve leakage and degraded ECCS systems that requires emergency Depressurization on low reactor water level.

Initial Conditions: Rx at 85% CTP, power ascension in progress, Suppression pool temperature is somewhat elevated due to weeping SRV's.

INOPERABLE Equipment
APRM 'F' is INOP due to a failed power supply card
ESF 12 Transformer is tagged out of service for maintenance
CCW Pump 'C' is tagged out of service for pump seal replacement
RHR 'C' Pump is tagged out of service for motor oil replacement
Appropriate clearances and LCOs are written.

Turnover: Continue to bring the plant to full power per IOI-2. Place RHR 'B' in Suppression Pool Cooling. There is severe weather reported in the Tensas Parish area.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP)	Startup RHR 'B' in Suppression Pool Cooling (SRO must declare LPCI 'B' Inop. (SOI 04-1-01-E12-1) (Tech Specs)
2		R (RO)	Raise Reactor power at least 5% using Recirc Flow Control.
3		C (RO)	Respond to Recirc Pump 'B' trip. (ONEP 05-1-02-III-3)
4	MS094	I (RO)	Respond to a failure of the Seal Steam Controller.

			(Alarm Response Instructions)

Scenario **4 BACK UP 1** (Continued)

Event No.	Malf. No.	Event Type*	Event Description
5	FW171B B21F065B_I RR063A .02	M (ALL)	Feedwater line "B" break in Drywell with check valve leakage - RO performs scram actions, attempts to isolate affected Feedwater line (valve will fail to close). - BOP recognizes that RCIC is unavailable (injects to affected Feedwater line)
6	RR040E 0 RR041E 1	I (BOP)	Failure of Division 1 ECCS to initiate
7	E22052	C (BOP)	HPCS pump trip on initiation
8	E12050B	C (BOP)	RHR B Pump trip
9		C (BOP)	LPCS Injection Valve failure to open

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

Critical Tasks

- Open at least five SRVs when RPV level drops to TAF.
- Manually initiate RHR A to restore RPV level above -167". Pump must be running before RPV pressure drops to 300 psig.
- If RPV Flooding is required, injects with RHR A to attempt to restore RPV pressure 57 psig above containment pressure.

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description:

Time	Position	Applicant's Actions or Behavior

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **4** Op-Test No.: **BACK UP 1**

Examiners: _____ Operators: _____
_____ _____
_____ _____

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

1. Raise power 5% with Recirc flow.
2. Startup one loop of Suppression Pool Cooling.
3. Implement Technical Specifications for LPCI inoperability of an RHR loop When it is running in suppression pool cooling.
4. Implement the AOP(ONEP) for a Recirc Pump trip.
5. Respond to a Seal Steam Controller failure.
6. Execute the EOPs for a Feedwater line break in the Drywell with check valve leakage and degraded ECCS systems that requires Emergency Depressurization on low reactor water level.

Initial Conditions: Rx at 85% CTP, power ascension in progress, Suppression pool temperature is somewhat elevated due to weeping SRV's.

INOPERABLE Equipment

APRM 'F' is INOP due to a failed power supply card
ESF 12 Transformer is tagged out of service for maintenance
CCW Pump 'C' is tagged out of service for pump seal replacement
RHR 'C' Pump is tagged out of service for motor oil replacement

Appropriate clearances and LCOs are written.

Turnover: Continue to bring the plant to full power per IOI-2. Place RHR 'B' in Suppression Pool Cooling. There is severe weather reported in the Tensas Parish area.

Scenario **4 BACK UP 1** (Continued)

Event No.	Malf. No.	Event Type*	10CFR55.45a	K/A	Event Description
1		N (BOP)	4, 8	219000 A4.01; A4.02; A4.07	Startup RHR 'B' in Suppression Pool Cooling (SRO must declare LPCI 'B' Inop. (SOI 04-1-01-E12-1) (Tech Specs)
2		R (RO)	1, 2, 4, 5, 6	202001 A4.04 202002 A4.08 2.2.2	Raise Reactor power at least 5% using Recirc Flow Control.
3	RR012B	C (RO)	3, 4, 5, 6	202001 A2.03	Respond to Recirc Pump 'B' trip ONEP. (ONEP 05-1-02-III-3)
4	MS094	I (RO)	3, 4, 8	295002 AK2.11 2.1.30	Respond to a failure of the Seal Steam Controller. (Alarm Response Instructions)
5	FW171B B21F065B_I RR063A .02	M (ALL)	3, 4, 5, 6, 7, 13	295031 EA1.0 203000 A3.08 241000 A4.06	Feedwater line "B" break in Drywell with check valve leakage - RO performs scram actions, attempts to isolate affected Feedwater line (valve will fail to close). - BOP recognizes that RCIC is unavailable
6	RR040E 0 RR041E 1	I (BOP)	3, 4, 5, 7	2.4.4 295024 EA1.0	Failure of Division 1 ECCS to initiate
7	E22052	C (BOP)	3, 5	295031 EA1.04 209002 A2.02	HPCS pump trip on initiation
8	E12050B	C (BOP)	3, 5	295031 EA1.01 203000 A2.02	RHR B Pump trip
9		C (BOP)	3, 5	295031 EA1.01 203000 A2.03	LPCS Injection Valve failure to open

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

Scenario **4 BACK UP 1** (Continued)

Critical Tasks

- Open at least five SRVs when RPV level drops to TAF.
- Manually initiate RHR A to restore RPV level above -167”. Pump must be running before RPV pressure drops to 300 psig.
- If RPV Flooding is required, injects with RHR A to attempt to restore RPV pressure 57 psig above containment pressure.

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description:

Time	Position	Applicant's Actions or Behavior

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **5** Op-Test No.: **BACK UP 2**

Examiners: _____ Operators: _____
_____ _____
_____ _____

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

1. Reduce power with Recirc flow
2. Respond to a Feedwater flow element failure that causes a feed pump trip.
3. Implement the AOP (ONEP) for a Feed Pump trip that places the plant in the immediate exit required region of the power-flow map.
4. Reduce power by inserting control rods.
5. Respond to overcurrent trip of a load center that causes entry into several LCO's
6. Execute the EOPs to respond to a feedwater line break in the Turbine Building with failure to scram that requires emergency depressurization on low reactor water level.
7. Recognize failure of SRV to close.

Initial Conditions: Rx at 100% CTP, earthquake just occurred.

INOPERABLE Equipment
APRM 'F' is INOP due to a failed power supply card
ESF 12 Transformer is tagged out of service for maintenance
CCW Pump 'C' is tagged out of service for pump seal replacement
RHR 'C' Pump is tagged out of service for motor oil replacement
RCIC tagged due to steam leak on turbine casing
Appropriate clearances and LCOs are written.

Turnover: Commence controlled plant shutdown (no apparent earthquake damage). There is severe weather reported in the Tensas Parish area.

Event No.	Malf. No.	Event Type*	Event Description
1		N (RO)	Discernible power reduction with recirc flow.
2	FTN21N 088B_B	I (RO)	Respond to RFPT trip due to flow instrument failure (minimum flow element for 'B' RFP)
3		R (RO)	Power reduction of >5% using control rods

Scenario **5 BACK UP 2** (Continued)

Event No.	Malf. No.	Event Type*	Event Description
4	R21142S	C (BOP)	ESF Load Center 15BA2 Trip
5	FW070A	M (ALL)	Feedwater line break in turbine building
6	C11164 0	M (ALL)	ATWS
7		C (BOP)	SRV (ADS) valve failure (valve stuck open)

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

Critical Tasks

- Insert control rods by scramming and/or driving.
- Inject SLC when suppression pool temperature cannot be maintained below 110°F.
- Terminate and prevent injection from Condensate / Feedwater, RHR A, RHR B, RHR C, and LPCS when RPV level cannot be maintained above -192”.
- Open at least five SRVs when when RPV level cannot be maintained above -192”.
- When reactor pressure drops to MARFP, restore injection from RHR A through E12-F053A, or RHR B through E12-F053B, and restore RPV level above -192”.

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description:

Time	Position	Applicant's Actions or Behavior

Facility: **GRAND GULF NUCLEAR STATION** Scenario No.: **5** Op-Test No.: **BACK UP 2**

Examiners: _____ Operators: _____
_____ _____
_____ _____

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

1. Reduce power with Recirc flow
2. Respond to a Feedwater flow element failure that causes a feed pump trip.
3. Implement the AOP (ONEP) for a Feed Pump trip that places the plant in the immediate exit required region of the power-flow map.
4. Reduce power by inserting control rods.
5. Respond to overcurrent trip of a load center that causes entry into several LCO's
6. Execute the EOPs to respond to a feedwater line break in the Turbine Building with failure to scram that requires emergency depressurization on low reactor water level.
7. Recognize failure of SRV to close.

Initial Conditions: Rx at 100% CTP, earthquake just occurred.

INOPERABLE Equipment

APRM 'F' is INOP due to a failed power supply card
ESF 12 Transformer is tagged out of service for maintenance
CCW Pump 'C' is tagged out of service for pump seal replacement
RHR 'C' Pump is tagged out of service for motor oil replacement
RCIC tagged due to steam leak on turbine casing

Appropriate clearances and LCOs are written.

Turnover: Commence controlled plant shutdown (no apparent earthquake damage). There is severe weather reported in the Tensas Parish area.

Scenario **5 BACK UP 2** (Continued)

Event No.	Malf. No.	Event Type*	10CFR55.45a	K/A	Event Description
1		N (RO)	2, 4, 5, 6	202001 A4.04 202002 A4.08 2.2.2	Discernible power reduction with recirc flow.
2	FTN21N0 088B_B	I (RO)	3, 4, 5, 6, 7	259001 A3.10 295001 AA1.01; AA1.05; AA2.01	Respond to RFPT trip due to flow instrument failure (minimum flow element for 'B' RFP)
3		R (RO)	2, 4, 5, 6	201005 A3.01; A3.02; A3.03; A4.01 2.2.2	Power reduction of >5% using control rods
4	R21142S	C (BOP)	3, 4, 8, 13	295003 AK2.04; AA1.01 2.1.31; 2.4.10; 2.4.48	ESF Load Center 15BA2 Trip
5	FW070A	M (ALL)	7, 13	295031 EA1.0 203000 A3.08 241000 A4.06	Feedwater line break in turbine building
6	C11164 0	M (ALL)	6, 8, 12, 13	295037 EA1.0; EA2.0 203000 A3.08 241000 A4.06	ATWS
7		C (BOP)	3, 4, 8	218000 A3.01; A4.01 239002 A2.03; A4.01	SRV (ADS) valve failure (valve stuck open)

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

Critical Tasks

- Insert control rods by scramming and/or driving.
- Inject SLC when suppression pool temperature cannot be maintained below 110°F.
- Terminate and prevent injection from Condensate / Feedwater, RHR A, RHR B, RHR C, and LPCS when RPV level cannot be maintained above -192".
- Open at least five SRVs when RPV level cannot be maintained above -192".

- When reactor pressure drops to MARFP, restore injection from RHR A through E12-F053A, or RHR B through E12-F053B, and restore RPV level above -192”.

Op-Test No.: _____ Scenario No.: _____ Event No.: _____ Page ___ of ___

Event Description:

Time	Position	Applicant's Actions or Behavior

E/APE#/NAME/SAFETY FUNCTION	K 1	K 2	K 3	A 1	A 2	G	TOPIC(S)	IMP	REC #	SRO/RO /BOTH	RELATED K/A	ORIGIN	NOTES:
295003 Partial or Complete Loss of AC Power/ VI CFR41.7					05		Given plant conditions and a Normal electrical line up and a loss of power determine the availability of safe shutdown systems and buses.	4.2		BOTH	AA1.03: 4.4	NEW	
295006 SCRAM / I CFR41.5/41.14		02					Describe the response of the Reactor Water Level Control System when reactor level reaches level 3 during a reactor scram.	3.8		BOTH		NEW	
295007 High Reactor Pressure / III CFR41.7					02		With the reactor operating at 100% power, describe the effects on the reactor if a MSIV isolates.	4.1		BOTH	AA2.03: 3.7	BANK	Reactor power and reactor level inside shroud
295009 Low Reactor Water Level / II CFR41.2/41.14	05						Identify the Reactor level required to provide adequate core circulation with no recirculation pumps or shutdown cooling system operating.	3.4		BOTH		NEW	
295010 High Drywell Pressure / V CFR41.9	02						Identify the Drywell pressure that will begin to clear the Drywell to Containment Suppression Pool Vents.	3.1		BOTH		NEW	
295013 High Suppression Pool Temp. / V CFR41.9/41.10				02			Given plant conditions, determine the cause of rising Suppression Pool Temperature.	3.9		BOTH		NEW	
295014 Inadvertent Reactivity Addition / I CFR41.1/41.2/41.6/43.6			01				With the reactor in startup conditions such that the reactor has dropped subcritical, what are the operator actions if a high worth control rod is withdrawn fully.	4.1		BOTH		NEW	Susquehanna reactivity event 7/98
295015 Incomplete SCRAM / I CFR41.6/43.6			01				Describe the method to be used to allow the insertion of control rods using RCIS during an ATWS.	3.7		BOTH		NEW	
295016 Control Room Abandonment / VII CFR41.7		01					Given conditions and the attachment to the Shutdown from Remote Shutdown Panels ONEP, determine the condition of selected equipment.	4.5		BOTH		NEW	
295017 High Offsite Release Rate / IX CFR41.11		04					Given plant conditions, determine the monitoring of a plant release.	3.3		BOTH			
295023 Refueling Accidents / VIII CFR41.2/41.10/41.12/43.4/43.5/43.6/43.7			01				Describe what personnel response to a fuel handling accident.	4.3		BOTH		NEW	
295024 High Drywell Pressure / V CFR41.9	01						Describe the implications of exceeding the Drywell maximum design pressure.	4.2		BOTH		NEW	
PAGE 1 TOTAL TIER 1 GROUP 1	3	3	3	1	2	0	PAGE TOTAL # QUESTIONS	12					

E/APE#/NAME/SAFETY FUNCTION	K 1	K 2	K 3	A 1	A 2	G	TOPIC(S)	IMP	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
295025 High Reactor Pressure / III CFR41.3/41.5/41.7/43.2		05					Describe the basis for the operation of Low-Low Set Safety Relief Valves.	4.2		BOTH		NEW	
295026 Suppression Pool High Water Temp. / V CFR41.7/41.9/41.10/41.14/43.5		01					Given plant conditions, determine when Suppression Pool cooling is no longer effective and alternate actions are required.	4.0		BOTH		NEW	
295027 High Containment Temperature / V CFR41.5/41.9/41.10/43.5		03					Given plant conditions, identify the cooling mechanisms for the Containment.	3.7		BOTH	EK2.01: 3.4	NEW	
295030 Low Suppression Pool Water Level / V CFR41.9						2. 4. 18	Describe the basis for the low level limit for emergency depressurization based on Suppression Pool Level.	3.6		BOTH	EK3.01: 4.1	NEW	° moved from A1 to Generics
295031 Reactor Low Water Level / II CFR41.7/41.10/43.5		12					Given plant conditions, determine the reaction of plant systems.	4.5		BOTH		NEW	
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / I CFR41.1/41.2/41.6/43.6				04			Determine when plant conditions allow the termination of Standby Liquid Control with multiple control rods stuck out and what is the basis.	4.5		BOTH	EK1.04: 3.6 EK1.05: 3.6 EA2.03: 4.4	NEW	
295038 High Offsite Release Rate / IX CFR41.7/43.4			03				Determine the conditions which would result in an isolation of the Control Room Ventilation System (with regard to High Radiation).	3.9		BOTH			
500000 High Containment Hydrogen Conc. / V CFR41.10/43.5						2. 1. 20	Given applicable SOP, graph, and plant conditions, determine the power settings for the Hydrogen Recombiners and time to full power.	4.2		BOTH	EA1.03: 3.2	NEW	←Generics
295025 High Reactor Pressure / III CFR41.5/43.2			02				Describe the basis for a Reactor Recirculation Pump trip on High Reactor Pressure.	4.1		BOTH		NEW	
PAGE 2 TOTAL TIER 1 GROUP 1	0	4	2	1	0	2	PAGE TOTAL # QUESTIONS	9					

←500000 random selection was EK1 which the only topic was Containment integrity, changed to generic 2.1.20 for execution of procedural steps concerning Hydrogen Recombiners.
° random selection for 295030 was A1, this was moved to generics based on number of questions in A1 and allowance for basis question on Suppression pool low level.

E/APE#/NAME/SAFETY FUNCTION	K 1	K 2	K 3	A 1	A 2	G	TOPIC(S)	IMP	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
295006 SCRAM / I CFR41.3/41.6				06			Describe the operation of the Control Rod Drive Hydraulic System following a reactor scram. (Scram NOT reset.)	3.6		**SRO**			
295010 High Drywell Pressure / V CFR41.9/41.10/43.5				07			Describe the methods to be used to lower Containment Pressure.	3.4		**SRO**			EP2 Att 14
295016 Control Room Abandonment / VII CFR41.5/41.10/43.5					06		Given parameters from the Remote Shutdown Panel indications, calculate cool down rate and determine acceptability.	3.5		**SRO**			
295015 Incomplete SCRAM / I CFR41.1/41.2/43.6	04						Describe the reaction of the core with an ATWS and lowering of reactor pressure.	3.8		**SRO**	AK1.02: 4.1		
295023 Refueling Accidents / VIII CFR41.2/41.6/43.6				03			Describe the requirements to ensure proper positioning of spent fuel bundles being moved on the Refueling Platform Grapple.	3.6		**SRO**			Damaged Fuel at RBS
PAGE 3 TOTAL TIER 1 GROUP 1	1	0	0	3	1	0	PAGE TOTAL # QUESTIONS	5					
PAGE 1 TOTAL TIER 1 GROUP 1	3	3	3	1	2	0	PAGE TOTAL # QUESTIONS	12					
PAGE 2 TOTAL TIER 1 GROUP 1	0	4	2	1	0	2	PAGE TOTAL # QUESTIONS	9					
K/A CATEGORY TOTALS:	4	7	5	5	3	2	TIER 1 GROUP 1 GROUP POINT TOTAL	26					

E/APE#/NAME/SAFETY FUNCTION	K 1	K 2	K 3	A 1	A 2	G	TOPIC(S)	IMP	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
295001 Partial or Complete Loss of Forced Core Flow Circulation / I & IV CFR41.10/41.5/43.5						2. 4. 11	Given plant conditions and the power to flow map, determine the actions to be taken.	3.6		BOTH	2.4.1: 4.6 AA2.01: 3.8	NEW	↑ moved AA2 to generics
295002 Loss of Main Condenser Vacuum / III CFR41.4/43.2			05				Describe the basis for the isolation of the Main Steam Isolation Valves on a loss of condenser vacuum.	3.4		BOTH	AA1.04: 3.4	NEW	→ moved AA1 to AK3
295004 Partial or Total Loss of DC Power / VI CFR41.7					04		Describe the effects of a loss of DC power on the actuation of ECCS systems.	3.3		BOTH		NEW	
295005 Main Turbine Generator Trip / III													
295008 High Reactor Water Level / II CFR41.4/41.5	01						Identify the affects of a High Reactor Water Level on the Main Turbine and Reactor Feed Pump Turbines.	3.2		BOTH		NEW	
295011 High Containment Temperature / V													
295012 High Drywell Temperature / V CFR41.9				02			Given plant conditions, determine the means of Drywell heat removal.	3.8		BOTH	AA1.01: 3.6	NEW	
295018 Partial or Total Loss of CCW / VIII CFR41.4	01						Determine the effects on the RWCU with a sustained loss of Plant Service Water.	3.6		BOTH		NEW	
295019 Partial or Total Loss of Inst. Air / VIII CFR41.4/41.9					02		Describe the response of the Instrument Air System and Primary and Secondary Containment Isolation Valves when header pressure drops to < 60 psig.	3.7		**SRO**	AK2.14: 3.2 AK2.09: 3.3	NEW	
295020 Inadvertent Cont. Isolation / V & VII CFR41.9				01			Given an isolation of the Containment, determine the ability to reset and unisolate plant systems.	3.6		BOTH			
295021 Loss of Shutdown Cooling / IV CFR41.5/43.5				04			Determine an approved method of Alternate decay heat removal.	3.7		BOTH		NEW	Mode 3
295022 Loss of CRD Pumps / I CFR41.5		04					Given plant conditions, describe the affects on reactor water level a loss of both Control Rod Drive Pumps.	2.7		BOTH		NEW	
PAGE 1 TOTAL TIER 1 GROUP 2	2	1	1	3	2	1	PAGE TOTAL # QUESTIONS	10					

↑ Moved from AA2 which was random selection to generics of AOP operations.
→ Moved from AA1 which was random selection to AK3 to determine basis for MSIV Isolation on loss of condenser vacuum AA1 topics tend to be lower level knowledge.

E/APE#/NAME/SAFETY FUNCTION	K 1	K 2	K 3	A 1	A 2	G	TOPIC(S)	IMP	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
295028 High Drywell Temperature / V CFR41.5/41.10/41.14/43.5					03		Given plant conditions, determine the availability of reactor water level indications.	3.9		BOTH	2.4.20: 4.0	MOD	↓ moved EA1 to EA2
295029 High Suppression Pool Water Level / V CFR41.7			01				Describe the basis for emergency depressurizaion of the reactor in a High Suppression Pool level condition.	3.9		BOTH	2.4.18: 3.6	NEW	
295032 High Secondary Containment Area Temperature / V CFR41.4/41.9/41.10/43.5				05			Given plant conditions, determine systems which should have isolated.	3.9		BOTH		NEW	
295033 High Secondary Containment Area Radiation Levels / IX CFR41.11/41.12/43.4			03				Given plant conditions, determine the plant response to conditions to control offsite radiation levels	3.9		BOTH		NEW	
295034 Secondary Containment Ventilation High Radiation / IX CFR41.4/41.11/41.12/41.13/43.4			03				Given the plant in a Refueling Outage with spent fuel being moved in the Auxiliary Building, describe the actions to be taken on Fuel Pool Sweep High Radiation Alarms.	4.4		**SRO**	EK3.01: 4.1 EK3.02: 4.1		Refueling SRO / Control Room Supervisor
295035 Secondary Containment High Differential Pressure / V CFR41.7/41.13/43.4			02				Describe the operation of the Secondary Containment Ventilation system and its response to an abnormal differential pressure in Secondary Containment.	3.5		BOTH			
295036 Secondary Containment High Sump/Area Water Level / V													
600000 Plant Fire On Site / VIII CFR41.10/43.5					17		Given a fire in the Main Control Room, describe the actions to be taken and what safe shutdown equipment is available for operation without suspect.	3.6		**SRO**		NEW	
PAGE 2 TOTAL TIER 1 GROUP 2	0	0	4	1	2	0	PAGE TOTAL # QUESTIONS	7					
PAGE 1 TOTAL TIER 1 GROUP 2	2	1	1	3	2	1	PAGE TOTAL # QUESTIONS	10					
K/A CATEGORY TOTALS:	2	1	5	4	4	1	TIER 1 GROUP 2 GROUP POINT TOTAL	17					

↓ random selection for 295012 and 295028 were both A1, 295028 was moved to A2 to allow different knowledge about High Drywell Temperature to be tested.

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
201005 RCIS CFR41.6								13				Given an alarm and RCIS indications, determine cause based on system lights.	3.8		BOTH		NEW	Group 1 or 2 rod drift
202002 Recirculation Flow Control CFR41.6									01			Given plant conditions, identify the plant response to a Recirc Flow Control Runback.	3.4		BOTH		NEW	Recirc Pumps in abnormal pump configuration
203000 RHR/LPCI: Injection Mode CFR41.7	02											Describe the response of the RHR (LPCI) system with a low Suppression Pool Water Level.	3.9		BOTH	K6.06: 3.9	NEW	
209001 LPCS CFR41.7/41.8										01		Describe the operation of the LPCS pump if the system is manually overridden with an initiation signal.	3.6		BOTH	A4.03: 3.6 A4.05: 3.6	NEW	
209002 HPCS CFR41.7/41.8		03										Describe the operation of HPCS following an initiation with a loss of AC power.	2.9		BOTH		MOD	NRC 98 – 19 conditions changed
211000 SLC CFR41.6/41.7				02								Describe the response of the SLC system when aligned for test tank operation and a SLC initiation signal.	3.2		BOTH	A2.06: 3.3 A2.07: 3.2	NEW	CR at GGNS
212000 RPS CFR41.6											2. 1. 31	Given plant conditions and indications, determine the probable cause (Blown Fuse causes partial reactor scram).	3.9		BOTH	2.4.48: 3.8 A2.19: 3.9	NEW	Blown fuse indication ″ moved from A3 to generics
215004 Source Range Monitor CFR41.6								02				Determine the ability to continue a reactor startup with more than one SRM in a Division failed.	3.7		BOTH	A2.03: 3.3	NEW	
PAGE 1 TOTAL TIER 2 GROUP 1	1	1	0	1	0	0	0	2	1	1	1	PAGE TOTAL # QUESTIONS	8					

″ 212000 random selection A3 moved to generics to encompass indication of failure which can cause an inadvertent partial reactor scram.

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
215005 APRM / LPRM CFR41.6								04				Determine RPS actuation and rod blocks based on LPRM inputs.	3.9		BOTH	A1.02: 4.0 A1.03: 3.6 A1.04: 4.1 A2.03: 3.8	MOD	NRC 98-15
216000 Nuclear Boiler Instrumentation CFR41.7				05								Describe logic required to initiate ECCS.	4.1		BOTH		NEW	
217000 RCIC CFR41.7/41.10					06							Given RCIC indications following a RCIC Turbine shutdown/trip, determine the cause of the RCIC failure.	2.7		BOTH	A4.02: 3.9	NEW	
218000 ADS CFR41.7/41.8					01							Given plant conditions, determine the operation of ADS.	3.8		BOTH	K4.02: 4.0 K4.03: 4.0	NEW	
223001 Primary CTMT and Auxiliaries CFR41.7										07		Given plant conditions, determine the operation of the Drywell vacuum breaker valves. (Drywell to Containment pressure differential)	4.1		BOTH	A4.06: 4.0 A3.02: 3.4	NEW	
223002 PCIS / Nuclear Steam Supply Shutoff CFR41.7/41.9/43.5											2. 4. 49	Given isolation conditions, evaluate indications to determine whether appropriate actions have occurred.	4.0		BOTH	A1.01: 3.5 A3.01: 3.4 2.4.46: 3.6 2.4.48: 3.8	NEW	K/A selected to generic
226001 RHR/LPCI: CTMT Spray Mode CFR41.9							01					Identify the response of Containment Pressure, Suppression Pool Temperature and level when Containment Spray is operated.	3.8		BOTH	A1.04: 3.6 A1.08: 3.4	NEW	①① moved from K5 to A1
239002 SRVs CFR41.3		01										Identify the power supply to the SRVs (ADS, Low-Low Set, and Normal).	3.2		BOTH		NEW	
241000 Reactor / Turbine Pressure Regulator CFR41.5						11						Describe the response of the plant with a failure of the Main Stop and Control Valves closed with the reactor at power.	3.4		BOTH	A1.01: 3.8 A1.02: 3.9 A1.07: 3.7	NEW	
PAGE 2 TOTALS TIER 2 GROUP 1	0	1	0	1	2	1	1	1	0	1	1	PAGE 2 TOTAL # QUESTIONS	9					

①① Random selection was K5, moved to A1 due to low discriminatory value of topic 226001 K5.

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
259002 Reactor Water Level Control CFR41.5				10								Describe the operation of the Feed water Level Control System on a failure of the Feed Flow input signal.	3.4		BOTH	K6.04: 3.1	NEW	
261000 SGTS CFR41.7/41.11	08											Describe the response of the radiation monitoring system for SBGT on an initiation of the system.	3.1		BOTH	K4.01: 3.8	NEW	
262001 AC Electrical Distribution CFR41.4								03				Determine the effects of a ground fault on BOP Transformer 12A.	4.3		BOTH		NEW	
264000 EDGs CFR41.8									01			Given plant conditions determine the type of operations expected for the Division I, II and HPCS Diesel Generators.	3.1		BOTH		MOD	
290001 Secondary CTMT CFR41.9						09						Describe the ability of Auxiliary Building Fire Protection system to be restored following an Auxiliary Building Isolation in conjunction with a loss of AC Power.	3.6		BOTH	A2.06: 4.0 286000 A2.09: 2.8	NEW	
203000 RHR/LPCI: Injection Mode CFR41.10									01			With the plant aligned for shutdown cooling and a LOCA, describe the alignment of RHR for LPCI operation.	3.7		BOTH	A3.08: 4.1 A4.02: 4.1 K4.01: 4.2		Manual realignment requirement.
PAGE 3 TOTALS TIER 2 GROUP 1	1	0	0	1	0	1	0	1	2	0	0	PAGE TOTAL # QUESTIONS	6					
PAGE 1 TOTALS TIER 2 GROUP 1	1	1	0	1	0	0	0	2	1	1	1	PAGE TOTAL # QUESTIONS	8					
PAGE 2 TOTALS TIER 2 GROUP 1	0	1	0	1	2	1	1	1	0	1	1	PAGE TOTAL # QUESTIONS	9					
K/A CATEGORY TOTALS:	2	2	0	3	2	2	1	4	3	2	2	TIER 2 GROUP 1 GROUP POINT TOTAL	23					

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
201001 CRD Hydraulic CFR41.6							03					Describe the reaction of the CRD System flows when moving a control rod.	2.8		BOTH		NEW	± moved from K5 to A1
202001 Recirculation CFR41.5/41.6/43.6					05							Given plant conditions, describe the operational sequence of an End of Cycle Recirculation Pump Trip.	3.6		BOTH		NEW	
204000 RWCU CFR41.7						07						Given plant conditions, determine operation of the Reactor Water Cleanup System upon an initiation of SLC B.	3.5		BOTH		NEW	
205000 Shutdown Cooling CFR41.7		02										Describe the affects of a loss of power to the RHR Shutdown Cooling suction valve with respect to interlocks with other RHR valves.	2.7		BOTH	A2.10: 2.9 A3.01: 3.1 K6.01: 3.4	NEW	
215003 IRM CFR41.6			01									Given IRM conditions, determine the status of control rod blocks and RPS.	4.0		BOTH	K3.03: 3.7 A1.03: 3.7 A1.04: 3.4		
219000 RHR /LPCI Suppression Pool Cooling Mode																		
234000 Fuel Handling Equipment																		
238003 MSIV Leakage Control																		
245000 Main Turbine Gen., and Auxiliaries CFR41.4						05						Given plant conditions, determine a condition that would result in a Turbine Generator trip.	2.9		BOTH	K6.04: 2.7	NEW	Generator Hydrogen Syst. Mod
259001 Reactor Feedwater CFR41.4			01									Given plant startup/heatup with Feedwater aligned to feed the reactor and a subsequent trip of the Reactor Feed pump, determine the action of Reactor Water Level.	3.9		BOTH		NEW	
PAGE 1 TOTAL TIER 2 GROUP 2	0	1	2	0	1	2	1	0	0	0	0	PAGE TOTAL # QUESTIONS	7					

± 201001 random selection K5 moved to A1 due to low importance factors of K5 topics and number of K5s selected on random.

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
262002 UPS (AC/DC)																		
263000 DC Electrical Distribution CFR41.4			02									Discern the effects of a loss of DC power on the operation of circuit breakers.	3.8		BOTH		BANK	①②moved from K4 to K3
271000 Offgas CFR41.7/41.13				04								Discuss the effects an emergency shutdown of the Hydrogen Water Chemistry System on the Offgas system.	3.6		BOTH	A2.07: 3.3 A1.14: 3.0 K6.06: 2.5	NEW	①③ moved from K3 to K4
272000 Radiation Monitoring																		
286000 Fire Protection CFR41.4									04			Given conditions requiring the initiation of fire deluge systems for a Containment Filter Train, describe how this in accomplished.	3.3		BOTH	2.1.30: 3.4 2.4.25: 3.4 2.4.27: 3.5	NEW	Manual system
290003 Control Room HVAC CFR41.4										01		Given plant conditions, determine the appropriate Control Room HVAC configuration.	3.2		BOTH	A3.01: 3.5		
300000 Instrument Air CFR41.4			02									Given a loss of Instrument Air, determine the affect on the ability to feed water to the reactor.	3.4		BOTH		NEW	①④Random selection replaced with K3.02.
400000 Component Cooling Water CFR41.4							02					Given plant conditions, determine the affects on CCW and alternate actions to be taken. (Loss of Plant Service Water No LOCA.)	2.8		BOTH	K1.01: 3.3	NEW	GGNS Event 10/99
PAGE 2 TOTALS	0	0	2	1	0	0	1	0	1	1	0	PAGE 3 TOTAL # QUESTIONS	6					
PAGE 1 TOTALS	0	1	2	0	1	2	1	0	0	0	0	PAGE 1 TOTAL # QUESTIONS	7					
K/A CATEGORY TOTALS:	0	1	4	1	1	2	2	0	1	1	0	TIER 2 GROUP 2 GROUP POINT TOTAL	13					

①④ Random selection was K2. K2 only had two topics which had no discretionary value. Replaced with K3.02 dealing with system interrelations.
①② Random selection was K4 that is N/A at GGNS, changed to K3.02.
①③Random selection was K3, changed to K4.04 to allow question concerning precautions about newly installed Hydrogen Water Chemistry System and hazards.

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	REC #	SRO/RO/ BOTH	RELATED K/A	ORIGIN	NOTES:
201003 Control Rod and Drive Mechanism																		
215001 Traversing In-core Probe																		
233000 Fuel Pool Cooling and Cleanup CFR41.4/41.10					01							Given plant conditions post refueling, determine the method of heat removal from the Fuel Pool Cooling Heat Exchangers.	2.7		BOTH		NEW	
239001 Main and Reheat Steam CFR41.4									01			Given plant conditions, determine the reaction of the Main and Reheat Steam system. (Isolation action)	4.1		BOTH		NEW	
256000 Reactor Condensate																		
268000 Radwaste CFR41.13/43.4											2. 4. 21	Given conditions and a Liquid Radwaste Discharge in Progress, determine whether a release will continue.	4.3		BOTH		MOD	①⑤ moved from K3 to generics
288000 Plant Ventilation																		
290002 Reactor Vessel Internals CFR41.3/41.14/43.2					05							Given plant conditions, determine the allowances based on the Pressure vs Temperature graph for the Reactor Vessel.	3.3		BOTH	2.1.25: 3.1	NEW	①⑥ moved from K1 to K5
K/A CATEGORY TOTALS:	0	0	0	0	2	0	0	0	1	0	1	TIER 2 GROUP 3 GROUP POINT TOTAL	4					

①⑤ Random selection was K3, Topic K3 does not have any discriminatory value, changed to generics 2.4.21.
①⑥ Random selection K1 topics sufficiently covered in other areas of examination, moved to K5 to evaluate Brittle Fracture Limits

CATEGORY	C1	C2	C3	C4	TOPIC(S)	IMP	REC #	SRO/RO /BOTH	RELATED K/A	ORIGIN	NOTES:
CONDUCT OF OPERATIONS – Minimum Shift Requirements CFR41.10	2.1.4				Given a situation that could cause shift staffing to drop below minimum, determine the proper course of action.	3.4		**SRO**		MOD	
CONDUCT OF OPERATIONS – Shift Relief and Turnover CFR41.10/43.2	2.1.3				Given a situation with oncoming shift personnel preparing to be relieved by personnel not physically capable of taking the shift, determine appropriate actions to be taken.	3.4		**SRO**		NEW	
CONDUCT OF OPERATIONS – Chemistry Limits CFR43.2	2.1.34				Given Chemistry report evaluate plant chemistry analysis, evaluate report against specifications and determine actions.	2.9		**SRO**		NEW	
CONDUCT OF OPERATIONS – Tech Specs CFR43.2	2.1.12				Given Technical Specifications and plant conditions, apply actions for a given situation. (SLC concentrations and conditions)	4.0		**SRO**	2.1.10: 3.9	NEW	
CONDUCT OF OPERATIONS – Responsibilities and Procedure usage CFR41.10	2.1.2				Given a situation which requires procedure changes to accomplish task, determine the actions to be taken.	4.0		**SRO**	2.1.20: 4.2 2.1.21: 3.2 2.1.23: 4.0	NEW	
EQUIPMENT CONTROL – Configuration Control CFR43.3		2.2.14			Describe the requirements for controlling changes to the plant configuration.	3.0		**SRO**		NEW	
EQUIPMENT CONTROL – Safety Limits CFR43.2		2.2.22			Given plant conditions, evaluate the status of plant Safety Limits.	4.1		**SRO**	2.1.10: 3.9 2.1.11: 3.8 2.1.33: 4.0	MOD	Apply new definition of Adequate Core Cooling
EQUIPMENT CONTROL – Maintenance Control CFR43.3		2.2.20			Given conditions requiring troubleshooting, describe the administrative requirements to allow maintenance.	3.3		**SRO**	2.2.19: 3.1	NEW	
EQUIPMENT CONTROL – Refueling CFR41.10/43.6/43.7		2.2.27			Describe the permissions required to deviate from the approved Fuel Movement Plan.	3.5		**SRO**	2.2.26: 3.7	NEW	
RADIATION CONTROL – Discharge Requirements CFR41.10/43.4			2.3.6		Describe the requirements to allow a batch release of a liquid tank to the environment.	3.1		**SRO**		NEW	
RADIATION CONTROL – Radioactive Material Transfer Requirements CFR41.10/43.4			2.3.3		Describe the Shift Superintendent responsibility for shipments of Radioactive material offsite.	2.9		**SRO**		NEW	HAZMAT PLAN
PAGE 1 TOTAL TIER 3	5	4	2	0	PAGE TOTAL # QUESTIONS	11					

CATEGORY	C1	C2	C3	C4	TOPIC(S)	IMP	REC #	SRO/RO /BOTH	RELATED K/A	ORIGIN	NOTES:
EMERGENCY PROCEDURES / PLAN – Fire Protection CFR41.10/43.2				2.4.25	Given a shift complement, determine which members are allowed to respond as the Fire Brigade and how many are required.	3.4		**SRO**	2.4.26: 3.3	NEW	
EMERGENCY PROCEDURES / PLAN – SAPs CFR41.10/43.5				2.4.4	Given plant conditions, identify the Severe Accident Procedures requiring implementation.	4.3		**SRO**		NEW	Entry into SAP
EMERGENCY PROCEDURES / PLAN – Protective Action Recommendations CFR41.10/43.5				2.4.44	Given plant conditions, determine appropriate Protective Action Recommendations to be sent to the State and Local officials.	4.0		**SRO**			
EMERGENCY PROCEDURES / PLAN – EOPs CFR41.10/43.5				2.4.16	Given an ATWS condition with the reactor below 1 % power and various number of control rods not fully inserted , determine alternate methods of control rod insertion with out RCIS.	4.0		**SRO**	2.4.21: 4.3 295015 AK2.01: 3.9 AA1.01: 3.9		EP2 Att 24
EMERGENCY PROCEDURES / PLAN – Offsite Notification Requirements CFR43.5				2.4.30	Given a situation involving a fire on site at the Hydrogen Bulk Storage Facility, determine EAL status and notification requirements to Offsite Agencies.	3.6		**SRO**	2.4.40: 4.0	MOD	Hydrogen Bulk Storage Fire at Fitzpatrick
EMERGENCY PROCEDURES / PLAN – SRO Responsibility CFR43.1/43.2/43.5				2.4.40	Given a situation, describe the responsibilities with respect to deviation from license conditions and approved procedures and determine actions.	4.0		**SRO**	2.4.47: 3.7 2.1.2: 4.0 2.4.49: 4.0	NEW	10 CFR 50.54 (x) Control Room Abandonment (terrorist attack)
PAGE 2 TOTAL TIER 3	0	0	0	6	PAGE TOTAL # QUESTIONS	6					
PAGE 1 TOTAL TIER 3	5	4	2	0	PAGE TOTAL # QUESTIONS	11					
K/A CATEGORY TOTALS:	5	4	2	6	TIER 3 GROUP POINT TOTAL	17					

BWR SRO EXAMINATION OUTLINE

Facility: **GRAND GULF NUCLEAR STATION** Date of Exam: **31 MARCH 2000**

TIER	GROUP	K/A CATEGORY POINTS											POINT TOTAL
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	
1. Emergency & Abnormal Plant Evolutions	1	4	7	5				5	3			2	26
	2	2	1	5				4	4			1	17
	TIER TOTAL	6	8	10				9	7			3	43
2. Plant Systems	1	2	2	0	3	2	2	1	4	3	2	2	23
	2	0	1	4	1	1	2	2	0	1	1	0	13
	3	0	0	0	0	2	0	0	0	1	0	1	4
	TIER TOTAL	2	3	4	4	5	4	3	4	5	3	3	40

3. Generic Knowledge & Abilities	CAT 1	CAT 2	CAT 3	CAT 4	
	5	4	2	6	17

Note:
1. Ensure that at least two topics from every K/A category are sampled within each tier (i.e., the "Tier Totals" in each K/A category shall not be less than two)
2. Actual point totals must match those specified in the table.
3. Select topics from many systems; avoid selecting more than two or three K/A topics from a given system unless they relate to plant specific priorities.
4. Systems / evolutions within each group are identified on the associated outline.
5. The shaded areas are not applicable to the category tier.
6.* The generic 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.
7. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings for the RO license level, and the point totals for each system and category. K/As below 2.5 should be justified on the basis of plant-specific priorities. Enter the tier totals for each category in the table above.

SYSTEM	WRITTEN	JPM	SCENARIOS
B13 RX VESSEL			
B21 VESSEL INST.			
B33-1 RECIRC			
B33-2 RECIRC FLO			
C11-1 CRD HYD			
C11-2 RCIS			
C11-3 CRD MECH			
C41 SLC			
C51 SRM			
C51 IRM			
C51 APRM			
C51 TIPS			
C61 REMOTE S/D			
C71 RPS			
D17/21 RAD MONIT			
E12 RHR S/D COOL			
E12 RHR LPCI			
E12 RHR SUPP POOL			
E12 RHR CTMT SPR			
E21 LPCS			
E22-1 HPCS			
E22-2 ADS			
SRVs			
E32/38 LEAK CONT			
E51 RCIC			
E61 COMB GAS			
F11 FUEL HANDLE			
G17/18 RADWASTE			
G33 RWCU			
G41 FPCCU			
L11 PLANT DC			
L62 UPS			
M41 CTMT			
T10 SEC CTMT			
M71 CTMT ISOL			
N11 MAIN STEAM			
N19 CONDENSATE			
N21 FEEDWATER			
N23/35 FW HEAT			
N30/40 TURB / GEN			
N32 TURBINE CONT			
N33 SEAL STEAM			
N62 COND AIR REM			
N64/65 OFFGAS			
N71 CIRC WATER			
P41 SSW			
P42 CCW			
P43 TBCW			
P44 PSW			
P53 INST AIR			
P64 FIRE PROTECT			
P75/81 DIESELS			
R20 AC POWER			
T41/42/46 AUX BLDG			
T48 SBGT			
VENTILATION			
Z17/51/77 CR AC			