Facility: Nine Mile Point 2 Date of Exam: 12/06/99 Exam Level: SRO

Tier	Group	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G *	Point Total
1. Emergency & Abnormal Plant Evolutions	1	4	5	4				5	3			5	26
	2	3	3	3				2	3			3	17
	Tier Totals	7	8	7				7	6			8	43
2. Plant Systems	1	3	1	2	2	1	2	3	2	2	2	3	23
	2	1	1	1	1	2	1	0	2	1	1	2	13
	3	0	1	0	0	0	1	0	1	0	0	1	4
	Tier Totals	4	3	3	3	3	4	3	5	3	3	6	40

3. Generic Knowledge and Abilities	Cat 1	Cat 2	Cat 3	Cat 4	Point Total
	5	4	4	4	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 SRO 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.

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295003 Partial or Complete Loss of AC Pwr / 6 *LER 99-010 PRA (IPE: AC Power Recovery)*		X					AK2.03 – Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: A.C. electrical distribution system.	3.9	1
295003 Partial or Complete Loss of AC Pwr / 6 *PRA (IPE: Divisional AC Failure)*						X	2.2.22 – Knowledge of limiting conditions for operations and safety limits.	4.1	1
295006 SCRAM / 1					X		AA2.06 – Ability to determine and/or interpret the following as they apply to SCRAM: Cause of Reactor Scram.	3.8	1
295006 SCRAM / 1		X					AK2.07 – Knowledge of the interrelations between SCRAM and the following: Reactor pressure control.	4.1	1
295007 High Reactor Pressure / 3			X				AK3.03 – Knowledge of the reasons for the following responses as they apply to High Reactor Pressure: RCIC operation; Plant Specific	3.5	1
295007 High Reactor Pressure / 3				X			AA1.04 – Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: Safety/relief valve operation: Plant-Specific.	4.1	1
295009 Low Reactor Water Level / 2						X	2.4.4 – Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	4.3	1
295010 High Drywell Pressure / 5				X			AA1.02 – Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Drywell floor and equipment drain sumps	3.6	1
295013 High Suppression Pool Temperature / 5			X				AK3.01 – Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Suppression pool cooling operation.	3.8	1
295014 Inadvertent Reactivity Addition / 1				X			AA1.02 – Ability to operate and/or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: Recirculation flow control system	3.8	1
295015 Incomplete SCRAM / 1		X					AK2.11 – Knowledge of the interrelations between INCOMPLETE SCRAM and the following: Instrument Air	3.7	1
295015 Incomplete SCRAM / 1				X			AA1.02 – Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: RPS	4.2	1
295016 Control Room Abandonment / 7						X	2.4.11 – Knowledge of abnormal condition procedure.	3.6	1
295017 High Off-Site Release Rate / 9					X		AA2.01 – † Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Off-site release rate: Plant-Specific	4.2	1

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295023 Refueling Accidents / 8				X			AA1.07 – Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Standby gas treatment/RFVS	3.6	1
295024 High Drywell Pressure / 5			X				EK3.04 – † Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Emergency depressurization	4.1	1
295025 High Reactor Pressure / 3	X						EK1.05 – † Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: Exceeding Safety Limits	4.7	1
295025 High Reactor Pressure / 3					X		EA2.04 – Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Suppression chamber pressure: Plant-Specific.	3.9	1
295026 Suppression Pool High Water Temperature / 5						X	2.2.12 – Knowledge of surveillance procedures.	3.4	1
295026 Suppression Pool High Water Temperature / 5	X						EK1.01 – Knowledge of the operational implications of he following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: Pump NPSH.	3.4	1
295030 Low Suppression Pool Water Level / 5			X				EK3.06 – Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: Reactor SCRAM.	3.8	1
295031 Reactor Low Water Level / 2		X					EK2.08 – Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Automatic depressurization system	4.3	1
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1						X	2.4.8 – Knowledge of how the event-based emergency/abnormal operating procedures are used in conjunction with the symptom-bases EOPs.	3.7	1
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	X						EK1.02 – Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Reactor water level effects on reactor power	4.3	1
295038 High Off-Site Release Rate / 9		X					EK2.05 - † Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Site emergency plan.	4.7	1
500000 High Containment Hydrogen Conc. / 5 *PRA (IPE: Containment Venting)*	X						EK1.01 – Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATIONS: Containment integrity	3.9	1
K/A Category Totals:	4	5	4	5	3	5	Group Point Total:		26

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1					X		AA2.01 – Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Power/flow map	3.8	1
295002 Loss of Main Condenser Vacuum / 3				X			AA1.07 – Ability to operate and/or monitor the following as they apply to a LOSS OF MAIN CONDENSER VACUUM: Condenser circulating water system	2.9	1
295004 Partial or Complete Loss of DC Pwr / 6 *PRA (IPE: Divisional DC/ Emergency DC Power)*	X						AK1.02 – Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Redundant D.C. power supplies: Plant –Specific	3.4	1
295005 Main Turbine Generator Trip / 3						X	2.1.33 – Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.	4.0	1
295008 High Reactor Water Level / 2	X						AK1.03 – Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL: Feed flow/steam flow mismatch	3.2	1
295012 High Drywell Temperature / 5		X					AK2.01 – Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell ventilation	3.5	1
295018 Partial or Complete Loss of CCW / 8			X				AK3.07 – Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Cross-connecting with backup systems	3.2	1
295019 Part. Or Comp. Loss of Inst. Air / 8 *PRA (IPE: Loss of Inst. Air)*						X	2.4.48 – Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.	3.8	1
295020 Inadvertent Cont. Isolation / 5					X		AA2.02 – Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION: Drywell containment temperature	3.4	1
295021 Loss of Shutdown Cooling / 4		X					AK2.04 – Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: Component cooling water systems: Plant-Specific	3.1	1
295028 High Drywell Temperature / 5	X						EK1.01 – Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Reactor water level measurement	3.7	1
295029 High Suppression Pool Water Level / 5			X				EK3.01 – Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL WATER LEVEL: Emergency depressurization	3.9	1
295033 High Sec. Cont. Area Rad. Levels / 9				X			EA1.01 – Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATIONS LEVELS: Area radiation monitoring system	4.0	1

ES-401	BWR SRO Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2								Form ES-401-1
E/APE # / Name / Safety Function	**K1**	**K2**	**K3**	**A1**	**A2**	**G**	**K/A Topic(s)**	**Imp.**	**Points**
295034 Sec. Cont. Ventilation High Rad. / 9						X	2.4.17 – Knowledge of EOP terms and definitions.	3.8	1
295035 Secondary Containment High Differential Pressure / 5			X				EK3.02 – Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Secondary containment ventilation response	3.5	1
295036 Secondary Containment High Sump/Area Water Level / 5					X		EA2.03 – Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Cause of the high water level	3.8	1
600000 Plant Fire On Site / 8		X					AK2.01 – Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors/ detectors and valves	2.7	1
K/A Category Point Totals:	3	3	3	2	3	3	Group Point Total:		17

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
202002 Recirculation Flow Control *DER 2-98-3370*									X			A3.01 – Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: flow control valve operation: BWR-5,6	3.4	1
203000 RHR/LPCI: Injection Mode					X							K5.01 – Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: INJECTION MODE (PLANT SPECIFIC): Testable check valve operation	2.9	1
209001 LPCS	X											K1.01 – Knowledge of the physical connections and/or cause-effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Condensate storage tank: Plant-Specific	3.1	1
209001 LPCS	X											K1.09 – Knowledge of the physical connections and/or cause-effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Nuclear boiler instrumentation	3.4	1
209002 HPCS *PRA (IPE: HPCS)*							X					A1.03 – Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) controls including: Reactor water level: BWR-5,6	3.7	1
211000 SLC				X								K4.03 – knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Keeping sodium pentaborate in solution	3.9	1
212000 RPS				X								K4.07 – Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Manual system activation (trip)	4.1	1
215004 SRM			X									K3.02 – Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: Reactor manual control: Plant-Specific	3.4	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
215005 APRM/LPRM	X											K1.14 – Knowledge of the physical connections and/or cause-effect relationships between AVERAGE POWER RANGE MONITOR/ LOCAL POWER RANGE MONITOR SYSTEM and the following: Reactor vessel	2.9	1
216000 Nuclear Boiler Instrumentation			X									K3.01 – Knowledge of the effect that a loss of malfunction of the NUCLEAR BOILER Instrumentation will have on following: Reactor Protection System	4.3	1
217000 RCIC *LER 99-010 PRA (IPE: RCIC)*						X						K6.03 – Knowledge of the effect that a loss of malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Suppression pool water supply	3.5	1
218000 ADS		X										K2.01 – Knowledge of electrical power supplies to the following: ADS logic	3.3	1
223001 Primary CTMT and Auxiliaries											X	2.4.45 – Ability to prioritize and interpret the significance of each annunciator or alarm.	3.6	1
223002 PCIS/Nuclear Steam Supply Shutoff								X				A2.01 – Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/ NUCLEAR STEAM SUPPLY SHUT-OFF; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. electrical distributions failures	3.5	1
226001 RHR/LPCI: Containment Spray System Mode *PRA (IPE: RHR)*							X					A1.05 – Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE controls including: System lineup	3.4	1
239002 Relief/Safety Valves										X		A4.06 – Ability to manually operate and/or monitor in the control room: Reactor water level	4.1	1
241000 Reactor/Turbine Pressure Regulator						X						K6.01 – Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR REGULATING SYSTEM: A.C. electrical power	2.9	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
259002 Reactor Water Level Control											X	2.1.6 – Ability to supervise and assume a management role during plant transients and upset conditions.	4.3	1
261000 SGTS										X		A4.07 – Ability to manually operate and/or monitor in the control room: System flow	3.2	1
262001 A.C. Electrical Distribution *PRA (IPE: LOSP-Blackout/ AC Power Recovery)*								X				A2.03 – Ability to (a) predict the impacts of the following on the A.C. ELECTRCIAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those abnormal conditions or operations: Loss of off-site power	4.3	1
264000 EDGs *PRA (IPE: Emerg AC Power/ Divisional AC Failures)*							X					A1.03 – Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: Operating voltages, currents, and temperatures	2.9	1
264000 EDGs									X			A3.06 – Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Cooling water system operations	3.2	1
290001 Secondary Containment											X	2.4.16 – Knowledge of EOP implementation hierarchy and coordination with other support procedures.	4.0	1
K/A Category Point Totals:	3	1	2	2	1	2	3	2	2	2	3	Group Point Total:		23

Plant Systems - Tier 2/Group 2

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201001 CRD Hydraulic		X										K2.05 – Knowledge of electrical power supplies to the following: Alternate rod insertion valve solenoids: Plant-Specific	4.5	1
201002 RMCS								X				A2.04 – Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Control rod block	3.1	1
204000 RWCS											X	2.4.48 – Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.	3.8	1
205000 Shutdown Cooling										X		A4.07 – Ability to manually operate and/or monitor in the control room: Reactor temperatures (moderator, vessel, flange)	3.7	1
214000 RPIS				X								K4.01 – Knowledge of ROD POSITION INFORMATION SYSTEM design feature(s) and/or interlocks which provide for the following: Reed switch locations	3.1	1
245000 Main Turbine Gen. And Auxiliaries					X							K5.02 – Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Turbine operation and limitations	3.1	1
259001 Reactor Feedwater *LER 99-010*											X	2.4.49 – Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.0	1
262002 UPS (AC/DC)						X						K6.01 – Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (AC/DC): A.C. electrical power	2.9	1
263000 DC Electrical Distribution			X									K3.03 – Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Systems with D.C. components (i.e. valves, motors, solenoids, etc.)	3.8	1

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

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
271000 Offgas									X			A3.02 – Ability to monitor automatic operations of the OFFGAS SYSTEM including: System flows	2.8	1
286000 Fire Protection								X				A2.06 – Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low fire main pressure: Plant-Specific	3.2	1
290003 Control Room HVAC	X											K1.04 – Knowledge of the physical connections and/or cause-effect relationships between CONTROL ROOM HVAC and the following: Nuclear steam supply shut off system (NSSSS/PCIS): Plant-Specific	3.3	1
300000 Instrument Air					X							K5.13 – Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Filters	2.9	1
K/A Category Point Totals:	1	1	1	1	2	1	0	2	1	1	2	Group Point Total:		13

Plant Systems - Tier 2/Group 3

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201003 Control Rod and Drive Mechanism						X						K6.01 – Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD AND DRIVE MECHANISM: Control rod drive hydraulic system	3.3	1
233000 Fuel Pool Cooling and Cleanup											X	2.1.14 – Knowledge of system status criteria which require the notification of plant personnel.	3.3	1
239001 Main and Reheat Steam *PRA (IPE: MSIV Closure)*		X										K2.01 – Knowledge of electrical power supplies to the following: Main steam isolation valve solenoids	3.3	1
290002 Reactor Vessel Internals								X				A2.04 - Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Excessive heatup/cooldown rate	4.1	1
K/A Category Point Totals:	0	1	0	0	0	1	0	1	0	0	1	Group Point Total:		4

Plant-Specific Priorities

System / Topic	Recommended Replacement for...	Reason	Points
Plant Specific Priorities coincided with randomly selected K/As.			
Plant-Specific Priority Total (limit 10):			

Category	K/A#	Topic	Imp.	Points
Conduct of Operations	2.1.4	Knowledge of shift staffing requirements.	3.4	1
	2.1.17	Ability to make accurate, clear and concise verbal reports	3.6	1
	2.1.16	Ability to operate plant phone, paging system, and two-way radio.	2.8	1
	2.1.20	Ability to execute procedure steps.	4.2	1
	2.1.12	Ability to apply technical specifications for a system. *PRA (IPE: Service Water)*	4.0	1
	Total			5
Equipment Control	2.2.26	Knowledge of refueling administrative requirements.	3.7	1
	2.2.17	Knowledge of the process for managing maintenance activities during power operations. *LER 99-010/ SOER 98-01*	3.5	1
	2.2.6	Knowledge of the process for making changes in procedures as described in the safety analysis report.	3.3	1
	2.2.23	Ability to track limiting conditions for operations.	3.8	1
	Total			4
Radiation Control	2.3.4	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	3.1	1
	2.3.11	Ability to control radiation releases.	3.2	1
	2.3.9	Knowledge of the process for performing a containment purge. *PRA (IPE: Cont. Vent.)*	3.4	1
	2.3.1	Knowledge of 10 CFR 20 and related facility radiation control requirements.	3.0	1
	Total			4

Category	K/A#	Topic	Imp.	Points
Emergency Procedures/Plan	2.4.1	Knowledge of EOP entry conditions and immediate action steps.	4.6	1
	2.4.32	Knowledge of operator response to a loss of all annunciators.	3.5	1
	2.4.19	Knowledge of EOP layout, symbols, and icons	3.7	1
	2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control 2. Core cooling and heat removal 3. Reactor coolant system integrity 4. Containment conditions 5. Radioactivity release control.	4.3	1
	Total			4
Tier 3 Point Total (RO/SRO)				17

Facility: Nine Mile Point 2
Date of Exam: 12/06/99 Exam Level: RO

Tier	Group	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G *	Point Total
1. Emergency & Abnormal Plant Evolutions	1	3	2	2				3	1			2	13
	2	3	3	4				3	3			3	19
	3	0	1	1				1	1			0	4
	Tier Totals	6	6	7				7	5			5	36
2. Plant Systems	1	3	3	3	2	2	3	3	1	3	2	3	28
	2	2	1	2	1	2	2	1	2	2	2	2	19
	3	1	0	0	1	0	0	1	1	0	0	0	4
	Tier Totals	6	4	5	4	4	5	5	4	5	4	5	51

3. Generic Knowledge and Abilities	Cat 1	Cat 2	Cat 3	Cat 4	
	3	3	4	3	13

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.

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295005 Main Turbine Generator Trip / 3						X	2.1.33 – Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications.	3.4	1
295006 SCRAM / 1					X		AA2.06 – Ability to determine and/or interpret the following as they apply to SCRAM: Cause of Reactor Scram. 3.7	3.5	1
295007 High Reactor Pressure / 3			X				AK3.03 – Knowledge of the reasons for the following responses as they apply to High Reactor Pressure: RCIC operation; Plant Specific	3.4	1
295009 Low Reactor Water Level / 2						X	2.4.4 – Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	4.0	1
295010 High Drywell Pressure / 5				X			AA1.02 – Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Drywell floor and equipment drain sumps	3.6	1
295014 Inadvertent Reactivity Addition / 1				X			AA1.02 – Ability to operate and/or monitor the following as they apply to INADVERTENT REACTIVITY ADDITION: Recirculation flow control system	3.6	1
295015 Incomplete SCRAM / 1		X					AK2.11 – Knowledge of the interrelations between INCOMPLETE SCRAM and the following: Instrument Air	3.5	1
295015 Incomplete SCRAM / 1				X			AA1.02 – Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: RPS	4.0	1
295024 High Drywell Pressure / 5			X				EK3.04 – † Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Emergency depressurization	3.7	1
295025 High Reactor Pressure / 3	X						EK1.05 – † Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE: Exceeding Safety Limits	4.4	1
295031 Reactor Low Water Level / 2		X					EK2.08 – Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Automatic depressurization system.	4.2	1
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	X						EK1.02 – Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSALE OR UNKNOWN: Reactor water level effects on reactor power	4.1	1
500000 High Containment Hydrogen Conc. / 5 *PRA (IPE: Containment Venting)*	X						EK1.01 – Knowledge of the operational implications of the following concepts as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATIONS: Containment integrity	3.3	1
K/A Category Totals:	3	2	2	3	1	2	Group Point Total:		13

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

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4					X		AA2.01 – Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Power/flow map	3.5	1
295002 Loss of Main Condenser Vacuum / 3				X			AA1.07 – Ability to operate and/or monitor the following as they apply to a LOSS OF MAIN CONDENSER VACUUM: Condenser circulating water system	3.1	1
295003 Partial or Complete Loss of AC Pwr / 6 *LER 99-010; PRA (IPE: AC Power Recovery)*		X					AK2.03 – Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: A.C. electrical distribution system	3.7	1
295004 Partial or Complete Loss of DC Pwr / 6 *PRA (IPE: Divisional D.C.)*	X						AK1.02 – Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Redundant D.C. power supplies: Plant –Specific	3.2	1
295008 High Reactor Water Level / 2	X						AK1.03 – Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL: Feed flow/steam flow mismatch	3.2	1
295012 High Drywell Temperature / 5		X					AK2.01 – Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Drywell ventilation	3.4	1
295013 High Suppression Pool Temp. / 5			X				AK3.01 – Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Suppression pool cooling operation	3.6	1
295016 Control Room Abandonment / 7				X			AA1.03 – Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT: RPIS	3.0	1
295017 High Off-site Release Rate / 9					X		AA2.01 – † Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: Off-site release rate: Plant Specific	2.9	1
295018 Partial or Complete Loss of CCW / 8			X				AK3.07 – Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: Cross-connecting with backup systems	3.1	1
295019 Part. Or Comp. Loss of Inst. Air / 8 *PRA (IPE: Loss of Inst. Air)*						X	2.4.48 – Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.	3.5	1
295020 Inadvertent Cont. Isolation / 5 & 7						X	2.4.11 – Knowledge of abnormal condition procedures.	3.4	1
295022 Loss of CRD Pumps / 1					X		AA2.02 – Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS: CRD system status.	3.3	1
295028 High Drywell Temperature / 5	X						EK1.01 – Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Reactor water level measurement	3.5	1

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295030 Low Suppression Pool Water Level / 5			X				EK3.06 – Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: Reactor SCRAM	3.6	1
295033 High Sec. Cont. Area Rad. Levels / 9			X				EK3.04 – Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS: Personnel evacuation	4.0	1
295034 Sec. Cont. Ventilation High Rad. / 9						X	2.4.17 – Knowledge of EOP terms and definitions.	3.1	1
295038 High Off-site Release Rate / 9		X					EK2.05 – † Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE: Site emergency plan	3.7	1
600000 Plant Fire On Site / 8				X			AA1.08 – Ability to operate and/or monitor the following as they apply to PLANT FIRE ON SITE: Fire fighting equipment used on each class of fire	2.6	1
K/A Category Point Totals:	3	3	4	3	3	3	Group Point Total:		19

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295021 Loss of Shutdown Cooling / 4		X					AK2.04 – Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: Component cooling water systems: Plant Specific	3.0	1
295023 Refueling Accidents / 8				X			AA1.07 – Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Standby gas treatment/FRVS	3.6	1
295035 Secondary Containment High Differential Pressure / 5			X				EK3.02 – Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: Secondary containment ventilation response.	3.3	1
295036 Secondary Containment High Sump/Area Water Level / 5					X		EA2.03 – Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL: Cause of the high water level	3.4	1
K/A Category Point Totals:	0	1	1	1	1	0	Group Point Total:		4

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201001 CRD Hydraulic System		X										K2.05 – Knowledge of electrical power supplies to the following: Alternate rod insertion valve solenoids: Plant-Specific	4.5	1
201001 CRD Hydraulic System							X					A1.03 – Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM controls including: CRD system flow	2.9	1
201002 RMCS								X				A2.04 – Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Control rod block.	3.2	1
202002 Recirculation Flow Control *DER 2-98-3370*									X			A3.01 – Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: flow control valve operation: BWR-5,6	3.6	1
203000 RHR/LPCI: Injection Mode					X							K5.01 – Knowledge of the operational implications of the following concepts as they apply to RHR/LPCI: INJECTION MODE (PLANT SPECIFIC): Testable check valve operation	2.7	1
209001 LPCS	X											K1.01 – Knowledge of the physical connections and/or cause-effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Condensate storage tank: Plant-Specific	3.1	1
209001 LPCS	X											K1.09 – Knowledge of the physical connections and/or cause-effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Nuclear boiler instrumentation	3.2	1
209002 HPCS *PRA (IPE: HPCS)*							X					A1.03 – Ability to predict and/or monitor changes in parameters associated with operating the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) controls including: Reactor water level: BWR-5,6	3.7	1
211000 SLC				X								K4.03 – knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Keeping sodium pentaborate in solution	3.8	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
212000 RPS				X								K4.07 – Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Manual system activation (trip)	4.1	1
215003 IRM					X							K5.03 – Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM: Changing detector position	3.0	1
215004 SRM			X									K3.02 – Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: Reactor manual control: Plant-Specific	3.4	1
215004 SRM		X										K2.01 – Knowledge of electrical power supplies to the following: SRM channels/detectors	2.6	1
215005 APRM/LPRM	X											K1.14 – Knowledge of the physical connections and/or cause-effect relationships between AVERAGE POWER RANGE MONITOR/ LOCAL POWER RANGE MONITOR SYSTEM and the following: Reactor vessel	2.8	1
216000 Nuclear Boiler Instrumentation									X			A3.01 – Ability to monitor automatic operations of the NUCLEAR BOILER Instrumentation including: Relationship between meter/recorder readings and actual parameter values: Plant-Specific	3.4	1
216000 Nuclear Boiler Instrumentation			X									K3.01 – Knowledge of the effect that a loss of malfunction of the NUCLEAR BOILER Instrumentation will have on following: Reactor Protection System	4.0	1
217000 RCIC *LER 99-010 PRA (IPE: RCIC)*						X						K6.03 – Knowledge of the effect that a loss of malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Suppression pool water supply	3.5	1
217000 RCIC *LER 99-010 PRA (IPE: RCIC)*										X		A4.09 – Ability to manually operate and/or monitor in the control room: System pressure	3.7	1
218000 ADS		X										K2.01 – Knowledge of electrical power supplies to the following: ADS logic	3.1	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
223001 Primary CTMT and Auxiliaries						X						K6.01 – Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES: Drywell cooling	3.6	1
223001 Primary CTMT and Auxiliaries											X	2.4.45 – Ability to prioritize and interpret the significance of each annunciator or alarm.	3.3	1
223002 PCIS/Nuclear Steam Supply Shutoff											X	2.1.32 – Ability to explain and apply system limits and precautions.	3.4	1
241000 Reactor/Turbine Pressure Regulator						X						K6.01 – Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR REGULATING SYSTEM: A.C. electrical power	2.8	1
259001 Reactor Feedwater *LER 99-010*											X	2.4.49 – Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.0	1
259002 Reactor Water Level Control			X									K3.05 – Knowledge of the effect that a loss of malfunction of the REACTOR WATER LEVEL CONTROL SYSTEM will have on following: Recirculation flow control system	2.8	1
261000 SGTS										X		A4.07 – Ability to manually operate and/or monitor in the control room: System flow	3.1	1
264000 EDGs *PRA (IPE: Emergency AC Power)*							X					A1.03 – Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: Operating voltages, currents, and temperatures	2.8	1
264000 EDGs									X			A3.06 – Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Cooling water system operations	3.1	1
K/A Category Point Totals:	3	3	3	2	2	3	3	1	3	2	3	Group Point Total:		28

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201003 Control Rod and Drive Mechanism						X						K6.01 – Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD AND DRIVE MECHANISM: Control rod drive hydraulic system	3.3	1
202001 Recirculation										X		A4.11 – Ability to manually operate and/or monitor in the control room: Seal pressures: Plant-Specific	3.2	1
204000 RWCS	X											K1.05 – Knowledge of the physical connections and/or cause-effect relationships between REACTOR WATER CLEANUP SYSTEM and the following: Plant air systems	2.7	1
214000 RPIS				X								K4.01 – Knowledge of ROD POSITION INFORMATION SYSTEM design feature(s) and/or interlocks which provide for the following: Reed switch locations	3.0	1
215002 RBM									X			A3.05 – Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including: Back panel meters and indicating lights: BWR-3, 4, 5	3.2	1
219000 RHR/LPCI: Torus/Pool Cooling Mode *PRA (IPE: RHR)*			X									K3.01 – Knowledge of the effect that a loss or malfunction of the RHR/LPCI: TORUS/ SUPPRESSION POOL COOLING will have the following: Suppression pool temperature control	3.9	1
239001 Main and Reheat Steam *PRA (IPE: MSIV Closure)*		X										K2.01 – Knowledge of electrical power supplies to the following: Main steam isolation valve solenoids	3.2	1
239001 Main and Reheat Steam											X	2.2.2 – Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.	4.0	1
245000 Main Turbine Gen. And Auxiliaries					X							K5.02 – Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS: Turbine operation and limitations	2.8	1
256000 Reactor Condensate										X		A4.10 – Ability to manually operate and/or monitor in the control room: Feedwater temperature	3.2	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
262001 AC Electrical Distribution *PRA (IPE: LOSP-Blackout/ AC Power Recovery)*								X				A2.03 – Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of offsite power	3.9	1
262002 UPS (AC/DC)						X						K6.01 – Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (AC/DC): A.C. electrical power	2.7	1
263000 DC Electrical Distribution			X									K3.03 – Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Systems with D.C. components (i.e. valves, motors, solenoids, etc.)	3.4	1
271000 Offgas									X			A3.02 – Ability to monitor automatic operations of the OFFGAS SYSTEM including: System flows	2.9	1
272000 Radiation Monitoring											X	2.4.46 – Ability to verify that the alarms are consistent with the plant conditions.	3.5	1
286000 Fire Protection								X				A2.06 – Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low fire main pressure: Plant-Specific	3.1	1
290001 Secondary CTMT	X											K1.02 – Knowledge of the physical connections and/or cause-effect relationships between SECONDARY CONTAINMENT and the following: Primary containment system: Plant-Specific	3.4	1
290003 Control Room HVAC							X					A1.05 – Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROOM HVAC controls including: Radiation monitoring (control room)	3.2	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
300000 Instrument Air					X							K5.13 – Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Filters	2.9	1
K/A Category Point Totals:	2	1	2	1	2	2	1	2	2	2	2	Group Point Total:		19

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
215001 Traversing In-Core Probe				X								K4.01 – Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following: Primary containment isolation: Mark I&II (Not-BWR1)	3.4	1
233000 Fuel Pool Cooling and Cleanup	X											K1.15 – Knowledge of the physical connections and/or cause-effect relationships between FUEL POOL COOLING AND CLEAN-UP and the following: Storage pools	2.9	1
234000 Fuel Handling Equipment							X					A1.01 – Ability to predict and/or monitor changes in parameters associated with operating the FUEL HANDLING EQUIPMENT controls including: Spent fuel pool level	3.1	1
290002 Reactor Vessel Internals								X				A2.04 – Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Excessive heatup/cooldown rate	3.7	1
K/A Category Point Totals:	1	0	0	1	0	0	1	1	0	0	0	Group Point Total:		4

Plant-Specific Priorities

System / Topic	Recommended Replacement for...	Reason	Points
Plant-Specific Priorities coincided with randomly selected KA's.			
Plant-Specific Priority Total: (limit 10)			

Category	K/A#	Topic	Imp.	Points
Conduct of Operations	2.1.17	Ability to make accurate, clear and concise verbal reports	3.5	1
	2.1.16	Ability to operate plant phone, paging system, and two-way radio.	2.9	1
	2.1.4 2.1.20	Ability to execute procedure steps.	4.3	1
	Total			3
Equipment Control	2.2.30	Knowledge of RO duties in the control room during fuel handling such as alarms from fuel handling area/ communication with fuel storage facility/ systems operated from the control room in support of fueling operations/ and supporting instrumentation.	3.5	1
	2.2.23	Ability to track limiting conditions for operations.	2.6.	1
	2.2.1	Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.	3.7	1
	Total			3
Radiation Control	2.3.2	Knowledge of facility ALARA program.	2.5	1
	2.3.11	Ability to control radiation releases.	2.7	1
	2.3.9	Knowledge of the process for performing a containment purge. *PRA (IPE: Cont. Vent)*	2.5	1
	2.3.1	Knowledge of 10 CFR 20 and related facility radiation control requirements.	2.6	1
	Total			4
Emergency Procedures/Plan	2.4.32	Knowledge of operator response to a loss of all annunciators.	3.3	1
	2.4.19	Knowledge of EOP layout, symbols, and icons	2.7	1
	2.4.21	Knowledge of the parameters and logic used to assess the status of safety functions including: 1. Reactivity control 2. Core cooling and heat removal 3. Reactor coolant system integrity 4. Containment conditions 5. Radioactivity release control.	37	1
	Total			3
Tier 3 Point Total (RO/SRO)				13

Facility: Nine Mile Point # 2	Date of Examination: ~~12/06/99~~ 2/14/00
Examination Level (circle one): **SRO**	Operating Test Number: Cat A Test 1

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Plant Parameter Verification	JPM: (New) Water Chemistry Operating Limits Determination (SRO ONLY). K/A 2.1.33, 2.1.34
	Shift Turnover	Question: 1. What are the requirements for maintaining an active license (Fill a Technical Specification required on-shift position)? K/A 2.1.1, 2.1.4
		Question: 2. After assuming the shift as the ASSS, what are the elements to be discussed at the shift brief? K/A 2.1.1, 2.1.3
A.2	Piping and Instrument Drawings	Question: 1. Using the PIDs, trace the Fire Protection Water flow path from the motor driven fire water pump 2FPW-P2, to the RPV using RHS Train A. 2RHS*MOV24A is available for injection. Where necessary, add EOP equipment to be used. K/A 2.1.24 *PRA (IPE: Fire Water – RHR Crosstie)*
		Question: 2. How do you verify that a PID is up to date and what is required to use it as a working copy? K/A 2.1.21
A.3	Radiation Work Permits	JPM (New) Review the attached RWP for task performance (GAP-RPP-02) K/A 2.3.7, 2.3.4, 2.3.10
A.4	Emergency Classification	JPM: (New) Emergency Plan classification of each SRO candidates scenario (to be administered after each scenario). K/A 2.4.29, 2.4.41

ES-301-1

Facility: Nine Mile Point # 2

Examination Level (circle one): **SRO**

Date of Examination: ~~12/06/99~~ 2/15/06

Operating Test Number: Cat A Test 2

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Startup Requirements	Question 1. During a reactor startup using control rod sequence A2UP, the ATC RO notices that at the completion of RWM Step 3, control rod 26-07 is at position 02. The RO reports that he failed to move the rod to position 04 when positioning it. Classify the reactivity management event. K/A 2.2.1, 2.2.35,
		Question 2. A reactor startup is in progress using control rod sequence A2UP. RWM Step 4 was just completed. Prior to and through the completion of RWM step 5, what reactivity controls must be in place? 2.1.2, 2.2.1, 2.2.36
	Security	Question 1. How are vital area keys in the Control Room controlled? K/A 2.1.2, 2.1.13
		Question 2. What are the responsibilities of Nine Mile Point employee who checks out a vital area key for a temporary job in the plant? K/A 2.1.2, 2.1.13
A.2	Surveillance Testing	Question: 1. On 12/14/99 at 0000 hrs it is discovered that N2-OSP-RHR-Q@004, RHR SYSTEM LOOP A PUMP & VALVE OPERABILITY TEST AND ASME XI PRESSURE TEST, was performed on 9/1/99 at 0000 hrs. What actions are required if the test cannot be performed within the next 48 hours? K/A 2.1.12, 2.2.12,
		Question 2. During a refueling outage, the 2DER*MOV120, EQUIP DRAINS OUTBD ISOL VLV, is scheduled to have its disk and seat replaced. Following completion of the work, what testing is required? K/A 2.1.12, 2.1.28, 2.1.33, 2.2.18, 2.2.21, 2.2.24
A.3	Radiation Monitoring	Question 1. When may the SSS waive a pre-job ALARA review? K/A 2.3.2
		Question 2. What installed radiation monitoring equipment is required on the refueling floor for a core offload (excluding personal radiation monitoring)? K/A 2.3.5
A.4	Emergency Classification	JPM: (New) Emergency Plan classification of each SRO candidates scenario (to be administered after each scenario). K/A 2.4.29, 2.4.41

Facility: Nine Mile Point # 2 Date of Examination: ~~12/06/99~~ 2/15/00

Examination Level (circle one): **RO** Operating Test Number: Cat A Test 1

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Shift Turnover	Question: 1.What are the requirements for maintaining an active license (Fill a Technical Specification required on-shift position)? K/A 2.1.4
		Question: 2. Following 4 days off you work day-shift (12 hour shifts) for five consecutive days, Thursday through Monday. You are called Monday night and asked to come in and work 12 hours on Tuesday day-shift. Determine if it is acceptable to work Tuesday including why or why not? K/A 2.1.1
	Start Up Requirements	Question: 1. Given SRM readings from N2-OP-101A (pg 6) and marked up Rod Sequence Pull Sheet, describe the rod movement restrictions that apply. K/A 2.2.1, 2.2.2, 2.2.35
		Question: 2. A reactor startup is in progress using Startup Control Sequence A2UP; currently performing step 9. Control rod 34-55 was just withdrawn to position 18 and the reactor is declared critical. The doubling time is 40 seconds. What actions are required? K/A 2.1.23, 2.2.1, 2.2.2
A.2	Piping and Instrument Drawings	Question: 1. Using the PIDs, trace the Fire Protection Water flow path from the motor driven fire water pump 2FPW-P2, to the RPV using RHS Train A. 2RHS*MOV24A is available for injection. Where necessary, add EOP equipment to be used. K/A 2.1.24 *PRA (IPE: Fire Water – RHR Crosstie)*
		Question: 2. How do you verify that a PID is up to date and what is required to use it as a working copy? K/A 2.1.24
A.3	Radiation Work Permits	JPM (New) Review the attached RWP for task performance (GAP-RPP-02) K/A 2.3.7, 2.3.4, 2.3.10
A.4	Emergency Classification	Question 1. The station is currently at an ALERT due to an ATWS. You are performing the actions to vent the scram air header when the STATION EVACUATION alarm is sounded and announcements for station evacuation are made. What are your actions in response to the Station Evacuation? K/A 2.4.12, 2.4.29, 2.4.34, 2.4.41
		Question 2. Following a Station Evacuation due to a LOCA, you are informed that two (2) maintenance workers are unaccounted for. The OSC is operational. What actions are required? K/A 2.4.39, 2.4.29, 2.4.42

Facility: Nine Mile Point # 2 Exam Level (circle one): RO / SRO		Date of Examination: 12/6/99 Operating Test No.: Simulator Day 5
B.1 Control Room Systems		
System / JPM Title	Type Code*	Safety Function
J1-1, O2-OPS-SJE-NEW, Manual Initiation of the Control Building Special Filter Train, K/A 290003, A4.01, 295038, EA1.07	N/S	9
J1-2, O2-OPS-SJE-264-2-04, Parallel Div I EDG with offsite (faulted) K/A 264000, A4.05 *PRA (IPE: AC Power Recovery)*	M/S/A	6
J1-3, O2-OPS-SJE-NEW, Add Water to the Suppression Pool via the HPCS System (faulted), K/A 223001, A1.08, A2.11, 295030, EA1.03	N/S/A	5
B.2 Facility Walk-Through		
* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA		

Facility: Nine Mile Point # 2
Exam Level (circle one): RO / SRO

Date of Examination: 2/14/00
Operating Test No.: Simulator Day 5

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
J1-1, O2-OPS-SJE-NEW, Manual Initiation of the Control Building Special Filter Train, K/A 290003, A4.01, 295038, EA1.07	N/S	9
J1-2, O2-OPS-SJE-264-2-04, Parallel Div I EDG with offsite (faulted) K/A 264000, A4.05 *PRA (IPE: AC Power Recovery)*	M/S/A	6
J1-3, O2-OPS-SJE-NEW, Add Water to the Suppression Pool via the HPCS System (faulted), K/A 223001, A1.08, A2.11, 295030, EA1.03	N/S/A	5

B.2 Facility Walk-Through

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

ES-301-2

Facility: Nine Mile Point = 2 Exam Level (circle one): RO / SRO	Date of Examination: 12/6/99 Operating Test No.: Simulator Day 6

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
J1-4, O2-OPS-SJE-NEW, Manually Initiate ADS (faulted), K/A 218000, A2.04, A4.01, A4.02 *PRA (IPE: Operator Depressurizes)*	N/S/A	3
J1-5, O2-OPS-SJE-NEW, Raising CRD Flow to the RPV After Shutdown During Emergency, K/A 295031, EA1.10	N/S	2
✱ J1-6, O2-OPS-SJE-NEW, Withdraw Control Rod – Uncoupled Rod (faulted), K/A 201003 A2.02, 201002 A1.02, A1.03	N/S/L/A	1
J1-7, O2-OPS-SJE-205-2-10, Reactor RHR B in Shutdown Cooling Following Shutdown for a Short Period, K/A 205000, A4.01, A4.03	D/S/L	4

B.2 Facility Walk-Through

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

Facility: Nine Mile Point # 2
Exam Level (circle one): RO / SRO

Date of Examination: 2/14/00
Operating Test No.: Simulator Day 6

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
J1-4, O2-OPS-SJE-NEW, Manually Initiate ADS (faulted), K/A 218000, A2.04, A4.01, A4.02 *PRA (IPE: Operator Depressurizes)*	N/S/A	3
J1-5, O2-OPS-SJE-NEW, Raising CRD Flow to the RPV After Shutdown During Emergency, K/A 295031, EA1.10	N/S	2
J1-6, O2-OPS-SJE-NEW, Withdraw Control Rod – Uncoupled Rod (faulted), K/A 201003 A2.02, 201002 A1.02, A1.03	N/S/L/A	1
J1-7, O2-OPS-SJE-205-2-10, Reactor RHR B in Shutdown Cooling Following Shutdown for a Short Period, K/A 205000, A4.01, A4.03	D/S/L	4

B.2 Facility Walk-Through

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

Facility: Nine Mile Point # 2 Exam Level (circle one): RO / SRO		Date of Examination: 12/6/99 Operating Test No.: Simulator Day 6
B.1 Control Room Systems		
System / JPM Title	Type Code*	Safety Function
J1-4, O2-OPS-SJE-NEW, Manually Initiate ADS (faulted), K/A 218000, A2.04, A4.01, A4.02 *PRA (IPE: Operator Depressurizes)*	N/S/A	3
J1-5, O2-OPS-SJE-NEW, Raising CRD Flow to the RPV After Shutdown During Emergency, K/A 295031, EA1.10	N/S	2
J1-6, O2-OPS-SJE-NEW, Withdraw Control Rod – Uncoupled Rod (faulted), K/A 201003 A2.02, 201002 A1.02, A1.03	N/S/L/A	1
J1-7, O2-OPS-SJE-205-2-10, Reactor RHR B in Shutdown Cooling Following Shutdown for a Short Period, K/A 205000, A4.01, A4.03	D/S/L	4
B.2 Facility Walk-Through		
* Type Codes: (D)irect from bank, (M)odified from bank, (N)ew, (A)lternate path, (C)ontrol room, (S)imulator, (L)ow-Power, (R)CA		

Facility: Nine Mile Point # 2 Exam Level (circle one): RO / SRO	Date of Examination: 12/6/99 Operating Test No.: Simulator Day 7

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
J2-4, O2-OPS-SJE-NEW, Lineup and Spray the Drywell following a LOCA (faulted), K/A 226001, A4.03	N/S/A	5
J2-5, O2-OPS-SJE-NEW, Vent the Reactor Pressure Vessel for Primary Containment flooding (faulted), K/A 295031, EA2.01, 239001, A2.03, A4.01, A4.02	N/S/A	4
J2-6, O2-OPS-SJE-201-2-22, Cooldown using Turbine Bypass Valves K/A 295025, EA1.02	D/S/L	3
J2-7, O2-OPS-SJE-NEW, Transfer Feedwater Control to High Pressure, Low Flow Control Valves, K/A 295002, A1.04, A4.01, A4.02, A4.03	N/S/L	2

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

Facility: Nine Mile Point # 2	Date of Examination: 02/14/00
Exam Level (circle one): RO / SRO	Operating Test No.: Simulator Day 7

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function
J2-4, O2-OPS-SJE-NEW, Lineup and Spray the Drywell following a LOCA (faulted), K/A 226001, A4.03	N/S/A	5
J2-5, O2-OPS-SJE-NEW, Vent the Reactor Pressure Vessel for Primary Containment flooding (faulted), K/A 295031, EA2.01, 239001, A2.03, A4.01, A4.02	N/S/A	4
J2-6, O2-OPS-SJE-201-2-22, Cooldown using Turbine Bypass Valves K/A 295025, EA1.02	D/S/L	3
J2-7, O2-OPS-SJE-MODIFIED, RCIC Turbine Reset, K/A 217000, A4.02	M/S/L	2

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

Facility: Nine Mile Point # 2
Exam Level (circle one): RO / SRO

Date of Examination: 2/14/00
Operating Test No.: Plant JPMs

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function

B.2 Facility Walk-Through

J1-8, 02-OPS-PJE-200-2-06, Defeat WCS Injection	D	3
J1-9, 02-OPS-PJE-200-2-69, Vent Control Rod overpiston volume	D/R	1
J1-10, 02-OPS-PJE-296-2-04, Manual operation of RCIC from RSP	D	2

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

Facility: Nine Mile Point # 2 Date of Examination: 12/6/99
Exam Level (circle one): RO / SRO Operating Test No.: Plant JPMs

DAY 4

B.1 Control Room Systems

System / JPM Title	Type Code*	Safety Function

B.2 Facility Walk-Through

J1-8, 02-OPS-PJE-200-2-06, Defeat WCS Injection	D	3
J1-9, 02-OPS-PJE-200-2-69, Vent Control Rod overpiston volume	D/R	1
J1-10, 02-OPS-PJE-296-2-04, Manual operation of RCIC from RSP	D	2

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

Scenario Outline

Nine Mile Point 2	Scenario No. Alternate	Operating Test No. 1

Examiners:	Candidates:

Objectives: Evaluate candidates ability to lower reactor power. Respond to a trip of a CRD Pump and failure of the on-line CRD flow controller. A Closed Cooling Water leak in the drywell. An un-isolatable steam line break in the drywell. The RHR Pump Suction Filters clog preventing the use of Drywell Sprays (unless SW is used). This may cause the containment to exceed PSP, prior to exceeding PSP the crew may elect to Alternate Depressurize. If PSP is exceeded it will require RPV Blowdown. Provides the ability evaluate normal, abnormal and emergency.

This scenario will be classified as an Alert (3.1.1)

Initial Conditions: 100% Power (IC-20), normal power operations, RHR Injection Valve under markup

Turnover: ,Normal operations. Assist maintenance as necessary with RHS*MOV24A

Event No.	Malf. No.	Type	Event Description
1	TC03A	I	(BOP) EHC Pressure Regulator Oscillation.
2		R	(RO) Power reduction to 90%
3		N	(BOP) Shift operating pressure regulators
4		C	(RO) Low suction trip of the operating CRD Pump.
5	RD14 A or B	I	(RO) Failure of the CRD flow controller. Valve fails closed.
6	CW06	C	(BOP) Closed cooling water (CCP) leak in the drywell, requires shutting down drywell coolers, entering N2-SOP-60
7	MS04	M	Steam line break in drywell, causing rising primary containment pressures and temperatures.

Scenario Outline

Nine Mile Point 2	Scenario No. 1	Operating Test No. 1

Examiners:	Candidates:

Objectives: Evaluate candidates ability to perform routine operating tasks, raise and lower reactor power. Respond to fuel failures and rising plant radiation levels. Failure of a main steam line radiation monitor. Clogging of condensate demineralizers resulting in a loss of feedwater. Failure of the RCIC flow controller and an steam line break in the secondary containment. Ability to execute normal, abnormal and emergency procedures, and insure compliance with Technical Specifications.

This scenario will be classified as a Site Area Emergency (3.4.1, 4.1.1, 4.2.1)

Initial Conditions: 90% Power (IC-20), normal power operations, power will be held at 90% until Reactor Engineering verifies acceptable thermal limits; HPCS has been out of service 16 hours and is inoperable due to injection valve CSH*MOV107 binding. Maintenance markup was issued, not expected back this shift.

Turnover: Hold power at 90% for Reactor Engineering and support maintenance recovery of HPCS. 14 day LCO, T.S. 3.5.1 for HPCS inop. Perform monthly Standby Gas Treatment surveillance.

Event No.	Malf. No.	Type	Event Description
1		N	(BOP) Perform monthly Standby Gas Treatment surveillance.
2	RX01	C	(RO) Fuel element failure resulting in raised off-gas and main steam line radiation, requiring power reduction (N2-SOP-17).
3		R	(RO) Reduce power with recirculation flow (N2-SOP-101D).
4	MS15D	I	(BOP/RO) Main steam line radiation monitor fails high, diagnose to determine instrument has failed, check T.S. (3.3.1)

Scenario Outline

Nine Mile Point 2	Scenario No. 3	Operating Test No. 1

Examiners:	Candidates:

Objectives: Evaluate candidates ability to raise and lower reactor power; respond to power monitoring instrument failures, generator failures, recirculation system problems, and electrical plant failures; maintain core coverage with a LOCA; control containment parameters, execute normal, abnormal and emergency procedures; ensure compliance with Technical Specifications.

This scenario will be classified as an Alert (3.1.1)

Initial Conditions: 85% Power (IC-20), Holding power for Reactor Engineering during a startup from a maintenance outage for unplanned main generator work, RCIC Tagged Out for coupling alignment.

Turnover: Transfer station NPS-SWG001 from Reserve Station Transformer to Normal Station Transformer, Technical Specification LCO in effect, 3.7.4, RCIC inoperable, 4 hours into 14 day LCO

Event No.	Malf. No.	Type	Event Description
1		N	(BOP) Transfer station NPS-SWG001 from Reserve Station Transformer to Normal Station Transformer
2	NM11	I	(RO) APRM Failure Upscale, Consult T.S., Bypass APRM
3	RR32	C	(RO) HPU A Oil Temp High, causing A FCV Lockup, restore and reset, check T.S. (3.4.1.3) *DER 2-99-3370*
* 4	EG04	C	(BOP) Main Generator Overheating, enter N2-OP-68, Sect. H, Off-Normal Operation. *PRA, (IPE: Turbine Trip)*
5		R	(RO) Lower power with recirc flow

* Removed Voltage Regulator Failure

Category C Simulator Evaluations

Day 1	SCENARIO #1 Candidate	Examiner
SRO	S3	E1
RO	R2	E2
BOP	R3	E3
SRO	S1	E1
RO	R1	E3
BOP	S2	E2
SRO	S7	E2
RO	S6	E1
BOP	R5	E3
SRO	S4	E1
RO	R4	E3
BOP	S5	E2

Day 2	SCENARIO #2 Candidate	Examiner
SRO	S6	E1
RO	R3	E3
BOP	R2	E2
SRO	S2	E2
RO	S1	E1
BOP	R1	E3
SRO	S5	E2
RO	S4	E1
BOP	R4	E3
SRO	Surrogate	
RO	S3	E1
BOP	S7	E2

Day 3	SCENARIO #3 Candidate	Examiner
SRO	Surrogate	
RO	R5	E2
BOP	S4	E1
SRO	Surrogate	
RO	S7	E2
BOP	S3	E1
SRO	Surrogate	
RO	S5	E2
BOP	S6	E1
SRO	Surrogate	
RO	S2	E3
BOP	S1	E1

Position	Candidate S1	S2	R1	S3	R2	R3	S4	S5	R4	S6	S7	R5
SRO	1	2		1			2	2		1	1	
*RO	2	3	1	2	1	2	1	3	1	2	3	3
BOP	3	1	2	3	2	1	3	1	2	3	2	1

*RO should be the Operator performing the "manipulation".
1,2 and 3 in the table indicate which scenario the candidate is in each position.

Operating Test Examiner Assignments

Examiner	Candidate			
E1	S1	S3	S4	S6
E2	S2	R2	S5	S7
E3	R1	R3	R4	R5

Each examiner is assigned 4 candidates.

Notes
1 scenario per day (same scenario run on 4 crews).
Each SRO candidate examined in 3 (1 SRO; 2 as RO)scenarios over 3 full days.
Each RO candidate is in 2 scenarios. Requires Surrogates.

Category B JPM's

Day 4 Plant JPM's

Examiner	Candidate	Plant JPM		
E1	S1	J1-8	J1-10	J1-9
E2	S2	J1-9	J1-8	J1-10
E3	R1	J1-10	J1-9	J1-8
E1	S3	J1-8	J1-10	J1-9
E2	R2	J1-9	J1-8	J1-10
E3	R3	J1-10	J1-9	J1-8

Examiner	Candidate	Plant JPM		
E1	S4	J1-8	J1-10	J1-9
E2	S5	J1-9	J1-8	J1-10
E3	R4	J1-10	J1-9	J1-8
E1	S6	J1-8	J1-10	J1-9
E2	S7	J1-9	J1-8	J1-10
E3	R5	J1-10	J1-9	J1-8

One full day of 3 plant JPM's
All candidates evaluated on same 3 plant JPM's
JPM's performed 3 at a time.

Day 5 Simulator JPM's

Examiner	Candidate	Sim JPM		
E1	S1	J1-1	J1-2	J1-3
E2	S2	J1-2	J1-3	J1-1
E3	R1	J1-3	J1-1	J1-2
E1	S3	J1-1	J1-2	J1-3
E2	R2	J1-2	J1-3	J1-1
E3	R3	J1-3	J1-1	J1-2

Examiner	Candidate	Sim JPM		
E1	S4	J1-1	J1-2	J1-3
E2	S5	J1-2	J1-3	J1-1
E3	R4	J1-3	J1-1	J1-2
E1	S6	J1-1	J1-2	J1-3
E2	S7	J1-2	J1-3	J1-1
E3	R5	J1-3	J1-1	J1-2

Notes One day of 3 simulator JPM's done simultaneously.
All 12 candidates complete same 3 sim JPM's
3 Days Total for simulator JPM's and Cat A Admin

Day 6 Simulator JPM's/Category A

	Examiner	Candidate	Sim JPM			
	E1	S1	J1-4	J1-5	E3	
	E2	S2	J1-5	J1-4	R1	
	E1	S1	J1-6	J1-7	Cat A	
2 hr	E2	S2	J1-7	J1-6	Test 1	2 hr
	E1	S3	J1-4	J1-5	E2	
	E3	R3	J1-5	J1-4	R2	
	E1	S3	J1-6	J1-7	Cat A	
2 hr	E3	R3	J1-7	J1-6	Test 1	2 hr
	E2	R2	J1-4	J1-5	E1	
	E3	R1	J1-5	J1-4	S1	
	E2	R2	J1-6	J1-7	Cat A	
2 hr	E3	R1	J1-7	J1-6	Test 1	2 hr
	E1	E2	E3	Cat A		
	S3	S2	R3	Test 1		

Notes 6 Candidates complete 4 sim jpm's and Category A

Day 7 Simulator JPM's/Category A

Examiner	Candidate	Sim JPM		
E1	S4	J2-4	J2-5	E3
E2	S5	J2-5	J2-4	R4
E1	S4	J2-6	J2-7	Cat A
E2	S5	J2-7	J2-6	Test 2
E2	S7	J2-4	J2-5	E1
E3	R5	J2-5	J2-4	S6
E2	S7	J2-6	J2-7	Cat A
E3	R5	J2-7	J2-6	Test 2
E1	S6	J2-4	J2-5	E2
E3	R4	J2-5	J2-4	S5
E1	S6	J2-6	J2-7	Cat A
E3	R4	J2-7	J2-6	Test 2
E1	E2	E3	Cat A	
S4	S7	R5	Test 2	

Notes 6 Candidates complete 4 sim jpm's and Category A

Revised

Scenario Outline

Nine Mile Point 2	Scenario No. 1	Operating Test No. 1

Examiners:	Candidates:

Objectives: Evaluate candidates ability to raise and lower power under normal and abnormal conditions; respond to fuel failures and rising plant radiation levels; failure of a main steam line radiation monitor; clogging of condensate demineralizers resulting in a loss of feedwater; failure of the RCIC flow controller; an un-isolatable steam line break in the secondary containment; ability to execute normal, abnormal and emergency procedures, and insure compliance with Technical Specifications.

This scenario will be classified as a Site Area Emergency (3.4.1, 4.1.1, 4.2.1)

Initial Conditions: 90% Power (IC-20), normal power operations, return to 100% power following a rod pattern adjustment; HPCS has been out of service 16 hours and is inoperable due to injection valve CSH*MOV107 binding. Maintenance markup issued, not expected back this shift.

Turnover: Continue the power ascension in accordance with N2-OP-101D and support maintenance recovery of HPCS. 14 day LCO, T.S. 3.5.1 for HPCS inop.

Event No.	Malf. No.	Type	Event Description
1		N	Raise power to 100% with recirculation flow.
2	RX01	C	(RO) Fuel element failure resulting in raised off-gas and main steam line radiation, requiring power reduction (N2-SOP-17).
3		R	(RO) Reduce power with recirculation flow (N2-SOP-101D).
4	MS15D	I	(BOP/RO) Main steam line radiation monitor fails high, diagnose to determine instrument has failed, check T.S. (3.3.1)

Scenario Outline

5	REM. FUNC.	C	(BOP/RO) Condensate demineralizers sequentially clog up requiring power reduction, placing more demineralizers in service and resulting in a loss of feedwater. *PRA (IPE: Loss of Feedwater)*
6	RC07	I	(BOP) RCIC flow controller fails high after initial operation, requiring manual control.
7		M	RCIC steam line break in the secondary containment, isolation valves fail to close, temperatures and radiation levels rise in secondary containment requiring RPV blowdown. *PRA (IPE: Emergency Depressurization), LER 99-010*

Scenario Outline

Nine Mile Point 2	Scenario No. 2	Operating Test No. 1

Examiners:	Candidates:

Objectives: Evaluate candidates ability to lower power under normal and abnormal conditions; respond to instrument and component failures encountered during surveillance testing and normal operations which require a Technical Specification 3.0.3 shutdown. Respond to a stuck open SRV; feedwater controller failure, EHC failure and failure of control rods to fully insert which results in an ATWS condition; lower RPV level to reduce power and control RPV pressure; execute normal, abnormal and emergency procedures; ensure compliance with Technical Specifications.

This scenario will be classified as a Site Area Emergency (2.2.2)

Initial Conditions: 100% Power (IC-20), normal power operations, SWP*P1C out of service.

Turnover: Continue normal power operations and return RCIC to operability following maintenance by performing of N2-OSP-ISC-Q@002, RCIC Pump and Valve Operability Test and System Integrity (completed through step B.2.21). SWP*P1C removed from service last shift for discharge strainer replacement..

Event No.	Malf. No.	Type	Event Description
1		N	(BOP) Perform N2-OSP-ISC-Q@002, RCIC Pump and Valve Operability Test and System Integrity
2	OVER-RIDES	I	(BOP) RHR flow instrument fails downscale, preventing Minimum Flow Valve (RHS*MOV4A (4B) from opening
3	AD05C	C	(BOP/RO) ADS Relief Valve opens, enter N2-SOP 34, pull fuses to close valve. Places plant in a condition requiring T.S. 3.0.3 shutdown *PRA (IPE: Inadvertent Open Safety Relief Valves)*
4		R	(RO) Reduce power with recirculation flow

Scenario Outline

5	FW14	I	(RO) Feedwater master controller fails low requiring manual control of feedwater. (N2-SOP-06) *PRA (IPE: Loss of Feedwater), LER 99-010*
6	OVER-RIDES	C	(BOP/RO) EHC system leak requiring power reduction per N2-SOP-101D
7	RD17Z	M	Control rods fail to fully insert, all turbine bypass valves fail closed as EHC pressure lowers from event 6. This requires the use of SRVs and lowering RPV level for pressure control. After control is established alternate methods must be used to scram the rods
8	RP08A RP08B	I	RRCS I and II 98 second timer failure, requiring manual SLC initiation

Scenario Outline

Nine Mile Point 2	Scenario No. 3	Operating Test No. 1
Examiners:		**Candidates:**

Objectives: Evaluate candidates ability to raise and lower power under normal and abnormal conditions; respond to power monitoring instrument failures, generator failures, recirculation system problems, and electrical plant failures; maintain core coverage with a LOCA; execute normal, abnormal and emergency procedures; ensure compliance with Technical Specifications.

This scenario will be classified as an Alert (3.1.1)

Initial Conditions: 75% Power (IC-16), continuation of a plant startup from a maintenance outage for unplanned main generator work, RCIC Tagged Out for coupling alignment.

Turnover: Continue the power ascension in accordance with N2-OP-101D, Technical Specification LCO in effect, 3.7.4, RCIC inoperable, 4 hours into 14 day LCO

Event No.	Malf. No.	Type	Event Description
1		N	(RO/BOP) Continue power ascension to 100% power./ Raise power with recirc flow
2	NM11	I	(RO) APRM Failure Upscale, Consult T.S., Bypass APRM
3	EG02	I	(BOP) Main Generator Automatic Voltage Regulator Fails High
4	RR32	C	(RO) HPU A Oil Temp High, causing A FCV Lockup, restore and reset, check T.S. (3.4.1.3) *DER 2-99-3370*
5	EG04	C	(BOP) Main Generator Overheating, enter N2-OP-68, Sect. H, Off-Normal Operation. *PRA, (IPE: Turbine Trip)*
6		R	(RO) Lower power with recirc flow

Scenario Outline

7	ED02A DG02A	C	Loss of Line 5, EDG 1 fails to start, resulting in a loss of power to bus 101, enter N2-SOP-3 and N2-SOP-11, requires a manual scram *PRA, (IPE: Divisional AC Failure) (IPE: Partial loss of Off-Site Power) (IPE: Operation of Service Water* *LER-99-010*
8	RR20	M	"A" FCV ruptures, HPCS is available to restore level, only 1 RHR pump is available for Suppression Pool cooling and Drywell Sprays.

Scenario Outline

Nine Mile Point 2	Scenario No. Alternate	Operating Test No. 1

Examiners:	Candidates:

Objectives: Evaluate candidates ability to lower power under normal and abnormal conditions; respond to a feedwater control system failure. After taking action and checking T.S. for the level transmitter An AO will contact the control room and notify them the operating CRD pump is making abnormal noise. This will require shifting CRD pumps. After the pump shift there will be a failure of the on-line CRD flow controller. After the controller failure there will be a closed cooling water system break in the drywell; an un-isolatable steam line break in the drywell. The RHR Pump Suction Filters clog preventing the use of Drywell Sprays (unless SW is used). This will cause the containment to exceed PSP, requiring RPV Blowdown. Providing the ability evaluate normal, abnormal and emergency procedure use, and insure compliance with Technical Specifications.

This scenario will be classified as an Alert (3.1.1)

Initial Conditions: 100% Power (IC-20), normal power operations

Turnover: , Lower power in preparation for a shutdown for scheduled refueling outage.

Event No.	Malf. No.	Type	Event Description
1		R	(RO/BOP) Lower power with recirculation flow.
2	FW30A	I	(RO) RPV level narrow range transmitter fails as is while in control, during power reduction. Enter N2-SOP-06, Tech. Specs. 3.3.9
3		N	(RO) Shift the operating CRD Pump.
4	RD14 A or B	I	(RO) Failure of the CRD flow controller. Valve fails closed.
5	CW06	C	(BOP) Closed cooling water (CCP) leak in the drywell, requires shutting down drywell coolers, entering N2-SOP-60
6	MS04	M	Steam line break in drywell, causing rising primary containment pressures and temperatures.

Scenario Outline

7	RH18 A/B/C	C	RHR Pump Suction Filters Clog. If operators continue to operate the pumps they will trip. This is a total loss of RHR and will cause the containment to exceed PSP, requiring RPV Blowdown.

NMPC NMP SIMULATOR SCENARIO

SCENARIO # 1- 1 **REV. 0** **No. of Pages: 21**

FUEL FAILURE, MAIN STEAM LINE RADIATION MONITOR FAILURE, LOSS OF FEED, RCIC CONTROLLER FAILURE, STEAM LEAK IN SECONDARY CONTAINMENT

PREPARER	G Bobka	DATE	1/31/00
VALIDATED	G Bobka with Ops Crew	DATE	1/31/00
CONFIGURATION CONTROL	NA Exam Security	DATE	NA
GEN SUPERVISOR OPS TRAINING	[signature]	DATE	2-1-00
OPERATIONS MANAGER UNIT 2	[signature] for DB	DATE	2-1-00

SCENARIO SUMMARY

Length: 60 minutes

SUMMARY

The scenario begins with the crew operating at 90% rated power. The crew will perform a Tech. Spec required functional test of Standby Gas Train A. Some accumulated material breaks loose in the reactor, reducing flow through some fuel bundles, then breaking up and passing down the steam lines. The first event is a small amount of fuel failure. Operators will lower power as Off-Gas and Main Steam radiation levels slowly rise. After power has been stabilized a Main Steam line Radiation Monitor will fail Hi Hi and fail to initiate a half scram requiring the crew to determine it's an instrument failure, manually insert a half scram and consult Tech. Specs. Material will begin to build up in the Condensate Demineralizers gbw ~~requiring a further power reduction~~ and eventually causing a trip of the feedwater system. The plant will be manually scrammed or automatically scram on level. RCIC will be initiated for level control but will experience a controller failure. RCIC may be operated in Manual to recover RPV level or the Feedwater System may be restarted. The fuel element failure will become worse and a steam leak will develop in the RCIC System resulting in high temperatures and radiation levels in the Reactor Building. The operators will attempt to isolate RCIC but the isolation valves will not work. The crew will be required to emergency depressurize to reduce the amount of energy released to the secondary containment.

EOPs exercised: RPV, SCC, RPV BLOWDOWN

Emergency Classification: SAE 3.4.1, 4.1.1, 4.2.1

Termination Criteria: RPV depressurized, RPV level stable

II. TURNOVER

SHIFT TURNOVER INFORMATION

REACTOR POWER	90%Rated
CORE LIFE	MOL
ROD LINE	>100%
SEQUENCE	A2DN
RWM STEP	29
SHIFT	DAYS/NIGHTS

A. Technical Specification LCOs in effect:
3.5.1, HPCS System inoperable 16 hours into 14 day LCO

B. Significant Problems/Abnormalities/Equipment Out of Service:
1. High Pressure Core Spray System inoperable due to Injection Valve CSH*MOV107 binding, maintenance marked up issued. Repairs are to be completed in about 24 hours.

C. Evolutions/Maintenance Scheduled for this Shift:
1. Hold power at 90% for Reactor Engineering to verify thermal limits then continue power ascension and support maintenance activities to restore HPCS
2. Perform N2-OSP-GTS-M001, GTS Functional Test for GTS Train A, for routine surveillance.

SHIFT COMPLEMENT

SRO	STRAHLEY
RO	RUSSELL
BOP	RESTUCCIO

NMPC NMP SIMULATOR SCENARIO

SCENARIO # 1-2 **REV. 0** **No. of Pages: 21**

FUEL FAILURE, MAIN STEAM LINE RADIATION MONITOR FAILURE, LOSS OF FEED, RCIC CONTROLLER FAILURE, STEAM LEAK IN SECONDARY CONTAINMENT

PREPARER	G Bobka	DATE	1/31/00
VALIDATED	G Bobka with Ops Crew	DATE	1/31/00
CONFIGURATION CONTROL	NA Exam Security	DATE	NA
GEN SUPERVISOR OPS TRAINING	[signature]	DATE	2-1-00
OPERATIONS MANAGER UNIT 2	[signature] for DB	DATE	2-1-00

SCENARIO SUMMARY

Length: 60 minutes

SUMMARY

The scenario begins with the crew operating at 90% rated power. The crew will perform a Tech. Spec required functional test of Standby Gas Train A. Some accumulated material breaks loose in the reactor, reducing flow through some fuel bundles, then breaking up and passing down the steam lines. The first event is a small amount of fuel failure. Operators will lower power as Off-Gas and Main Steam radiation levels slowly rise. After power has been stabilized a Main Steam line Radiation Monitor will fail Hi Hi and fail to initiate a half scram ~~requiring the crew to determine it's an instrument failure, manually insert a half scram and consult Tech. Specs. Material will begin to build up in the Condensate Demineralizers requiring a further power reduction and eventually causing a trip of the feedwater system~~. The plant will be manually scrammed or automatically scram on level. RCIC will be initiated for level control but will experience a controller failure. RCIC may be operated in Manual to recover RPV level or the Feedwater System may be restarted. The fuel element failure will become worse and a steam leak will develop in the RCIC System resulting in high temperatures and radiation levels in the Reactor Building. The operators will attempt to isolate RCIC but the isolation valves will not work. The crew will be required to emergency depressurize to reduce the amount of energy released to the secondary containment. gmw

EOPs exercised: RPV, SCC, RPV BLOWDOWN

Emergency Classification: SAE 3.4.1, 4.1.1, 4.2.1

Termination Criteria: RPV depressurized, RPV level stable

II. TURNOVER

SHIFT TURNOVER INFORMATION

REACTOR POWER	90%Rated
CORE LIFE	MOL
ROD LINE	>100%
SEQUENCE	A2DN
RWM STEP	29
SHIFT	DAYS/NIGHTS

A. Technical Specification LCOs in effect:
3.5.1, HPCS System inoperable 16 hours into 14 day LCO

B. Significant Problems/Abnormalities/Equipment Out of Service:
1. High Pressure Core Spray System inoperable due to Injection Valve CSH*MOV107 binding, maintenance marked up issued. Repairs are to be completed in about 24 hours.

C. Evolutions/Maintenance Scheduled for this Shift:
1. Hold power at 90% for Reactor Engineering to verify thermal limits then continue power ascension and support maintenance activities to restore HPCS
2. Perform N2-OSP-GTS-M001, GTS Functional Test for GTS Train A, for routine surveillance.

SHIFT COMPLEMENT

SRO	RICHARDSON
RO	JONES
BOP	ORZELL

SCENARIO # 1-3 **REV. 0** **No. of Pages: 21**

FUEL FAILURE, MAIN STEAM LINE RADIATION MONITOR FAILURE, LOSS OF FEED, RCIC CONTROLLER FAILURE, STEAM LEAK IN SECONDARY CONTAINMENT

PREPARER	G Bolton	DATE	1/31/00
VALIDATED	G Bolton with Ops Crew	DATE	1/31/00
CONFIGURATION CONTROL	NA Exam Security	DATE	NA
GEN SUPERVISOR OPS TRAINING	[signature]	DATE	2-1-00
OPERATIONS MANAGER UNIT 2	[signature] for DB	DATE	2-1-00

SCENARIO SUMMARY

Length: 60 minutes

SUMMARY

The scenario begins with the crew operating at 90% rated power. The crew will perform a Tech. Spec required functional test of Standby Gas Train A. Some accumulated material breaks loose in the reactor, reducing flow through some fuel bundles, then breaking up and passing down the steam lines. The first event is a small amount of fuel failure. Operators will lower power as Off-Gas and Main Steam radiation levels slowly rise. After power has been stabilized a Main Steam line Radiation Monitor will fail Hi Hi and fail to initiate a half scram requiring the crew to determine it's an instrument failure, manually insert a half scram and consult Tech. Specs. Material will begin to build up in the Condensate Demineralizers requiring a further power reduction and eventually causing a trip of the feedwater system. The plant will be manually scrammed or automatically scram on level. RCIC will be initiated for level control but will experience a controller failure. RCIC may be operated in Manual to recover RPV level or the Feedwater System may be restarted. The fuel element failure will become worse and a steam leak will develop in the RCIC System resulting in high temperatures and radiation levels in the Reactor Building. The operators will attempt to isolate RCIC but the isolation valves will not work. The crew will be required to emergency depressurize to reduce the amount of energy released to the secondary containment.

EOPs exercised: RPV, SCC, RPV BLOWDOWN

Emergency Classification: SAE 3.4.1, 4.1.1, 4.2.1

Termination Criteria: RPV depressurized, RPV level stable

II. TURNOVER

SHIFT TURNOVER INFORMATION

REACTOR POWER	90%Rated
CORE LIFE	MOL
ROD LINE	>100%
SEQUENCE	A2DN
RWM STEP	29
SHIFT	DAYS/NIGHTS

A. Technical Specification LCOs in effect:
   3.5.1, HPCS System inoperable 16 hours into 14 day LCO

B. Significant Problems/Abnormalities/Equipment Out of Service:
   1. High Pressure Core Spray System inoperable due to Injection Valve CSH*MOV107 binding, maintenance marked up issued. Repairs are to be completed in about 24 hours.

C. Evolutions/Maintenance Scheduled for this Shift:
   1. Hold power at 90% for Reactor Engineering to verify thermal limits then continue power ascension and support maintenance activities to restore HPCS
   2. Perform N2-OSP-GTS-M001, GTS Functional Test for GTS Train A, for routine surveillance.

SHIFT COMPLEMENT

SRO	CHWALEK
RO	DOWNS
BOP	FREGEAU

NMPC NMP SIMULATOR SCENARIO

SCENARIO # 2-1 **REV. 0** **No. of Pages: 24**

FAILURE TO SCRAM WITH LOSS OF EHC PRESSURE

PREPARER	G Bolton	DATE	1/31/00
VALIDATED	G Bolton with Ops Crew	DATE	1/31/00
CONFIGURATION CONTROL	NA Exam Security	DATE	NA
GEN SUPERVISOR OPS TRAINING	Steve R	DATE	2-1-00
OPERATIONS MANAGER UNIT 2	Matthew J Whitaker	DATE	2-1-00

SCENARIO SUMMARY

Length: 60 minutes

SUMMARY

While operating at rated power, the crew will prepare to perform RCIC full flow test surveillance. When RHR is initiated in Suppression Pool Cooling, for the test, the RHR flow Instrument will fail and the Minimum Flow Valve, MOV 4A(B), will cycle and NOT stay Closed when flow is established. This will make RHR Inoperative and the RCIC Test should be postponed while Tech. Specs. are checked.

After the Tech. Specs. are determined ADS/SRV*PSV137 opens due to a switch failure. Per SOP-34, fuses will be pulled. After the C and A fuses are pulled the SRV will shut. Again this will place the SRO in Tech. Specs. The loss of RHR and ADS will require a Tech. Specs. 3.0.3 Shutdown. When management is notified they will request the shutdown be started immediately. After conditions have stabilized the Feedwater Master Controller fails low, causing RPV level to lower. Feedwater Control must be placed in Manual. After conditions have stabilized the EHC System will develop a leak requiring a power reduction with Feedwater Control in Manual. After a power reduction, the EHC Pumps trip, and the reactor will either be manually scrammed or scram on high pressure.

When the reactor is scrammed the control rods fail to fully insert with a failure of the Redundant Reactivity Control System. The rods will not respond to manual scram signals until after the SDV is drained. Control Rods may be manually inserted. The operators enter and execute EOPs, RPV, PC and C5 as well as the appropriate off normal procedures.

EOPs Exercised: RPV, PCC, C5
Emergency Classification: SAE 2.2.2
Termination Criteria: RPV water level and pressure are under control. Suppression Pool temperature is stable or lowering. Actions have been taken or directed to insert rods in accordance with N2-EOP-6, Attachment 14.

II. TURNOVER

SHIFT TURNOVER INFORMATION

REACTOR POWER	100%
CORE LIFE	MOL
ROD LINE	>100%
SEQUENCE	A2DN
RWM SREP	29
SHIFT	DAYS/NIGHTS

A. Technical Specification LCOs in effect:
14 day LCO, Technical Specification 3.7.4, entered 3 hours ago when RCIC was declared inoperable per N2-OSP-ICS-Q002, RCIC Surveillance procedure.

B. Significant Problems/Abnormalities/Equipment Out of Service:
Service Water Pump 1C, SWP*P1C, removed from service last shift for discharge strainer replacement. Work has NOT been started yet.

C. Evolutions/Maintenance Scheduled for this Shift:
1. After assuming the shift, the SSS has directed the crew to start RHR "A" in Suppression Pool Cooling per N2-OP-31 to lower suppression pool temperature to 80°F prior to continuing the RCIC surveillance. [illegible] 2/15/00
2. Perform RCIC Full Flow Test, N2-OSP-ICS-Q@002 currently completed through step 8.3.1.
3. Continue power operations.

SHIFT COMPLEMENT

SSS	FREGEAU
RO	CHWALEK
BOP	DOWNS

Replaced Turnover sheet with Alex Herbi permission 2/15/00

NMPC NMP SIMULATOR SCENARIO

SCENARIO # 2-2 **REV. 0** **No. of Pages: 24**

FAILURE TO SCRAM WITH LOSS OF EHC PRESSURE

PREPARER	G Bottn	DATE	1/31/00
VALIDATED	G Bottn with Ops Crew	DATE	1/31/00
CONFIGURATION CONTROL	NA Exam Security	DATE	NA
GEN SUPERVISOR OPS TRAINING	Steve R	DATE	2-1-00
OPERATIONS MANAGER UNIT 2	Matthew J Weldecker	DATE	2-1-00

SCENARIO SUMMARY

Length: 60 minutes

SUMMARY

While operating at rated power, the crew will prepare to perform RCIC full flow test surveillance. When RHR is initiated in Suppression Pool Cooling, for the test, the RHR flow Instrument will fail and the Minimum Flow Valve, MOV 4A(B), will cycle and NOT stay Closed when flow is established. This will make RHR Inoperative and the RCIC Test should be postponed while Tech. Specs. are checked.

After the Tech. Specs. are determined ADS/SRV*PSV137 opens due to a switch failure. Per SOP-34, fuses will be pulled. After the C and A fuses are pulled the SRV will shut. Again this will place the SRO in Tech. Specs. The loss of RHR and ADS will require a Tech. Specs. 3.0.3 Shutdown. When management is notified they will request the shutdown be started immediately. After conditions have stabilized the Feedwater Master Controller fails low, causing RPV level to lower. Feedwater Control must be placed in Manual. After conditions have stabilized the EHC System will develop a leak requiring a power reduction with Feedwater Control in Manual. After a power reduction, the EHC Pumps trip, and the reactor will either be manually scrammed or scram on high pressure.

When the reactor is scrammed the control rods fail to fully insert with a failure of the Redundant Reactivity Control System. The rods will not respond to manual scram signals until after the SDV is drained. Control Rods may be manually inserted. The operators enter and execute EOPs, RPV, PC and C5 as well as the appropriate off normal procedures.

EOPs Exercised: RPV, PCC, C5
Emergency Classification: SAE 2.2.2
Termination Criteria: RPV water level and pressure are under control. Suppression Pool temperature is stable or lowering. Actions have been taken or directed to insert rods in accordance with N2-EOP-6, Attachment 14.

II. TURNOVER

SHIFT TURNOVER INFORMATION

REACTOR POWER	100%
CORE LIFE	MOL
ROD LINE	>100%
SEQUENCE	A2DN
RWM SREP	29
SHIFT	DAYS/NIGHTS

A. Technical Specification LCOs in effect:
14 day LCO, Technical Specification 3.7.4, entered 3 hours ago when RCIC was declared inoperable per N2-OSP-ICS-Q002, RCIC Surveillance procedure.

B. Significant Problems/Abnormalities/Equipment Out of Service:
Service Water Pump 1C, SWP*P1C, removed from service last shift for discharge strainer replacement. Work has NOT been started yet.

C. Evolutions/Maintenance Scheduled for this Shift:
1. After assuming the shift, the SSS has directed the crew to start RHR "A" in Suppression Pool Cooling per N2-OP-31 to lower suppression pool temperature to 80°F prior to continuing the RCIC surveillance.
2. Perform RCIC Full Flow Test, N2-OSP-ICS-Q@002 currently completed through step 8.3.1.
3. Continue power operations.

[initials] 2/1/00

SHIFT COMPLEMENT

SSS	RESTUCCIO
RO	STRAHLEY
BOP	RUSSELL

Replaced Turnover sheet with Herbi permission 2/1/00

NMPC NMP SIMULATOR SCENARIO

SCENARIO # 2-3 **REV. 0** **No. of Pages: 24**

FAILURE TO SCRAM WITH LOSS OF EHC PRESSURE

PREPARER	G Bottle	DATE	1/31/00
VALIDATED	G Bottle with Ops Crew	DATE	1/31/00
CONFIGURATION CONTROL	NA Exam Security	DATE	NA
GEN SUPERVISOR OPS TRAINING	Steve R	DATE	2-1-00
OPERATIONS MANAGER UNIT 2	Matthew J Whitaker	DATE	2-1-00

SCENARIO SUMMARY

Length: 60 minutes

SUMMARY

While operating at rated power, the crew will prepare to perform RCIC full flow test surveillance. When RHR is initiated in Suppression Pool Cooling, for the test, the RHR flow Instrument will fail and the Minimum Flow Valve, MOV 4A(B), will cycle and NOT stay Closed when flow is established. This will make RHR Inoperative and the RCIC Test should be postponed while Tech. Specs. are checked.

After the Tech. Specs. are determined ADS/SRV*PSV137 opens due to a switch failure. Per SOP-34, fuses will be pulled. After the C and A fuses are pulled the SRV will shut. Again this will place the SRO in Tech. Specs. The loss of RHR and ADS will require a Tech. Specs. 3.0.3 Shutdown. When management is notified they will request the shutdown be started immediately. After conditions have stabilized the Feedwater Master Controller fails low, causing RPV level to lower. Feedwater Control must be placed in Manual. After conditions have stabilized the EHC System will develop a leak requiring a power reduction with Feedwater Control in Manual. After a power reduction, the EHC Pumps trip, and the reactor will either be manually scrammed or scram on high pressure.

When the reactor is scrammed the control rods fail to fully insert with a failure of the Redundant Reactivity Control System. The rods will not respond to manual scram signals until after the SDV is drained. Control Rods may be manually inserted. The operators enter and execute EOPs, RPV, PC and C5 as well as the appropriate off normal procedures.

EOPs Exercised: RPV, PCC, C5
Emergency Classification: SAE 2.2.2
Termination Criteria: RPV water level and pressure are under control. Suppression Pool temperature is stable or lowering. Actions have been taken or directed to insert rods in accordance with N2-EOP-6, Attachment 14.

II. TURNOVER

SHIFT TURNOVER INFORMATION

REACTOR POWER	100%
CORE LIFE	MOL
ROD LINE	>100%
SEQUENCE	A2DN
RWM SREP	29
SHIFT	DAYS/NIGHTS

A. Technical Specification LCOs in effect:
14 day LCO, Technical Specification 3.7.4, entered 3 hours ago when RCIC was declared inoperable per N2-OSP-ICS-Q002, RCIC Surveillance procedure.

B. Significant Problems/Abnormalities/Equipment Out of Service:
Service Water Pump 1C, SWP*P1C, removed from service last shift for discharge strainer replacement. Work has NOT been started yet.

C. Evolutions/Maintenance Scheduled for this Shift:
1. After assuming the shift, the SSS has directed the crew to start RHR "A" in Suppression Pool Cooling per N2-OP-31 to lower suppression pool temperature to 80°F prior to continuing the RCIC surveillance.
2. Perform RCIC Full Flow Test, N2-OSP-ICS-Q@002 currently completed through step 8.3.1.
3. Continue power operations.

[illegible] 2/15/01

SHIFT COMPLEMENT

SSS	ORZELL
RO	RICHARDSON
BOP	JONES

Replaced Turnover sheet with Herbi permission 2/15/01

NMPC NMP SIMULATOR SCENARIO

SCENARIO #3 - 1 **REV. 0** **No. of Pages: 21**

LOCA WITH LOSS OF HIGH PRESSURE INJECTION

PREPARER	G Bobbin	DATE	1/31/00
VALIDATED	G Bobbin with Ops Crew	DATE	1/31/00
CONFIGURATION CONTROL	NA EXAM SECURITY	DATE	NA
GEN SUPERVISOR OPS TRAINING	Steve [illegible]	DATE	2-1-00
OPERATIONS MANAGER UNIT 2	Matthew J Wildecker	DATE	2-1-00

SCENARIO SUMMARY

Length: 60 minutes

The scenario begins with the plant at 85% power and continuing a power ascension following a forced outage to repair main generator seals. The crew will transfer station switchgear NPS-SWG001 from Reserve Station Transformer to Normal Station Transformer for a transformer outage. Crew will begin power rise and #2 APRM will fail upscale. After completing the diagnosis and taking the appropriate actions the oil cooler on the A Recirc HPU fails, the backup also fails causing a recirc FCV Lockup.

After taking action for the FCV lockup, the Main Generator will begin overheating requiring a power reduction. As power is reduced, Line 5 will be lost and EDG 1 will fail to start. This will require a manual scram. The Recirc FCV will rupture at this time (on a slow ramp) and Condensate Pump B will trip. All feeedwater will be lost. HPCS will initiate and restore RPV level. ~~RHR B must be used for Containment Spray~~. gHw

EOPs Exercised: RPV, PCC

Emergency Classification: Alert 3.1.1

Termination Criteria: RPV level restored, Primary Containment parameters under control.

II. TURNOVER

SHIFT TURNOVER INFORMATION

REACTOR POWER	85%
CORE LIFE	MOL
ROD LINE	>100%
SEQUENCE	A2UP
RWM STEP	29
SHIFT	DAYS/NIGHTS

A. Technical Specification LCOs in effect:

1. 3.7.4, RCIC inoperable, 4 hours into 14 day LCO

B. Significant Problems/Abnormalities/Equipment Out of Service:

1. RCIC is marked up for coupling alignment
2. NPS-SWG001 is being powered from Reserve Transformer A. Normal Station Transformer breaker has been returned to service.

C. Evolutions/Maintenance Scheduled for this Shift:

1. Remain at 85% power for Reactor Engineering
2. Support maintenance on RCIC System and return the Station to 100% power when directed
3. Transfer NPS-SWG001 from Reserve Station Transformer to Normal Station Transformer using normal operating procedures following breaker repair for the normal supply breaker.

SHIFT COMPLEMENT

SRO	SURROGATE
RO	RESTUCCIO
BOP	STRAHLEY

NMPC NMP SIMULATOR SCENARIO

SCENARIO #3-2 **REV. 0** **No. of Pages: 21**

LOCA WITH LOSS OF HIGH PRESSURE INJECTION

PREPARER	G Bobbin	DATE	1/31/00
VALIDATED	G Bobbin with Ops Crew	DATE	1/31/00
CONFIGURATION CONTROL	NA EXAM SECURITY	DATE	NA
GEN SUPERVISOR OPS TRAINING	Steve [signature]	DATE	2-1-00
OPERATIONS MANAGER UNIT 2	[signature]	DATE	2-1-00

SCENARIO SUMMARY

Length: 60 minutes

The scenario begins with the plant at 85% power and continuing a power ascension following a forced outage to repair main generator seals. The crew will transfer station switchgear NPS-SWG001 from Reserve Station Transformer to Normal Station Transformer for a transformer outage. Crew will begin power rise and #2 APRM will fail upscale. ~~After completing the diagnosis and taking the appropriate actions the oil cooler on the A Recirc HPU fails, the backup also fails causing a recirc FCV Lockup~~. gMu

~~After taking action for the FCV lockup,~~ The Main Generator will begin overheating requiring a power reduction. As power is reduced, Line 5 will be lost and EDG 1 will fail to start. This will require a manual scram. The Recirc FCV will rupture at this time (on a slow ramp) and Condensate Pump B will trip. All feeedwater will be lost. HPCS will initiate and restore RPV level. RHR B must be used for Containment Spray.

EOPs Exercised: RPV, PCC

Emergency Classification: Alert 3.1.1

Termination Criteria: RPV level restored, Primary Containment parameters under control.

II. TURNOVER

SHIFT TURNOVER INFORMATION

REACTOR POWER	85%
CORE LIFE	MOL
ROD LINE	>100%
SEQUENCE	A2UP
RWM STEP	29
SHIFT	DAYS/NIGHTS

A. Technical Specification LCOs in effect:

1. 3.7.4, RCIC inoperable, 4 hours into 14 day LCO

B. Significant Problems/Abnormalities/Equipment Out of Service:

1. RCIC is marked up for coupling alignment
2. NPS-SWG001 is being powered from Reserve Transformer A. Normal Station Transformer breaker has been returned to service.

C. Evolutions/Maintenance Scheduled for this Shift:

1. Remain at 85% power for Reactor Engineering
2. Support maintenance on RCIC System and return the Station to 100% power when directed
3. Transfer NPS-SWG001 from Reserve Station Transformer to Normal Station Transformer using normal operating procedures following breaker repair for the normal supply breaker.

SHIFT COMPLEMENT

SRO	SURROGATE
RO	ORZELL
BOP	RICHARDSON

NMPC NMP SIMULATOR SCENARIO

SCENARIO #3 - 3 **REV. 0** **No. of Pages: 21**

LOCA WITH LOSS OF HIGH PRESSURE INJECTION

PREPARER	G Bobka	DATE	1/31/00
VALIDATED	G Bobka with Ops Crew	DATE	1/31/00
CONFIGURATION CONTROL	NA EXAM SECURITY	DATE	NA
GEN SUPERVISOR OPS TRAINING	[signature]	DATE	2-1-00
OPERATIONS MANAGER UNIT 2	[signature]	DATE	2-1-00

SCENARIO SUMMARY

Length: 60 minutes

The scenario begins with the plant at 85% power and continuing a power ascension following a forced outage to repair main generator seals. The crew will transfer station switchgear NPS-SWG001 from Reserve Station Transformer to Normal Station Transformer for a transformer outage. Crew will begin power rise and #2 APRM will fail upscale. ~~After completing the diagnosis and taking the appropriate actions the oil cooler on the A Recirc HPU fails, the backup also fails causing a recirc FCV Lockup.~~ gHW

gHW ~~After taking action for the FCV lockup,~~ The Main Generator will begin overheating requiring a power reduction. As power is reduced, Line 5 will be lost and EDG 1 will fail to start. This will require a manual scram. The Recirc FCV will rupture at this time (on a slow ramp) and Condensate Pump B will trip. All feeedwater will be lost. HPCS will initiate and restore RPV level. RHR B must be used for Containment Spray.

EOPs Exercised: RPV, PCC

Emergency Classification: Alert 3.1.1

Termination Criteria: RPV level restored, Primary Containment parameters under control.

II. TURNOVER

SHIFT TURNOVER INFORMATION

REACTOR POWER	85%
CORE LIFE	MOL
ROD LINE	>100%
SEQUENCE	A2UP
RWM STEP	29
SHIFT	DAYS/NIGHTS

A. Technical Specification LCOs in effect:

1. 3.7.4, RCIC inoperable, 4 hours into 14 day LCO

B. Significant Problems/Abnormalities/Equipment Out of Service:

1. RCIC is marked up for coupling alignment
2. NPS-SWG001 is being powered from Reserve Transformer A. Normal Station Transformer breaker has been returned to service.

C. Evolutions/Maintenance Scheduled for this Shift:

1. Remain at 85% power for Reactor Engineering
2. Support maintenance on RCIC System and return the Station to 100% power when directed
3. Transfer NPS-SWG001 from Reserve Station Transformer to Normal Station Transformer using normal operating procedures following breaker repair for the normal supply breaker.

SHIFT COMPLEMENT

SRO	DOWNS
RO	FREGEAU
BOP	CHWALEK