INITIAL SUBMITTAL OF THE OUTLINE AND NRC COMMENTS

FOR THE PERRY INITIAL EXAMINATION - MARCH 2002

PERRY

INITIAL LICENSE EXAM

MARCH 5 THRU 13, 2002

Form ES-201-2, "Examination Outline Quality Checklist," ***along*** with the ***written examination and operating test outline(s)***

Facility: Date of Examination:

Item	Task Description	Initials a	Initials b*	Initials c#
1. WRITTEN	a. Verify that the outline(s) fit(s) the appropriate model per ES-401.	DPJ	RJS	DLP
	b. Assess whether the outline was systematically and randomly prepared in accordance with Section D.1 of ES-401 and whether all K/A categories are appropriately sampled.	DPJ	RJS	DLP
	c. Assess whether the outline over-emphasizes any systems, evolutions, or generic topics.	DPJ	RJS	DLP
	d. Assess whether the justifications for deselected or rejected K/A statements are appropriate.	DPJ	RJS	DLP
2. SIM	a. Using Form ES-301-5, verify that the proposed scenario sets cover the required number of normal evolutions, instrument and component failures, and major transients.	DPJ	RJS	DLP
	b. Assess whether there are enough scenario sets (and spares) to test the projected number and mix of applicants in accordance with the expected crew composition and rotation schedule without compromising exam integrity; ensure each applicant can be tested using at least one new or significantly modified scenario, that no scenarios are duplicated from the applicants' audit test(s)*, and scenarios will not be repeated over successive days.	DPJ	RJS	DLP
	c. To the extent possible, assess whether the outline(s) conform(s) with the qualitative and quantitative criteria specified on Form ES-301-4 and described in Appendix D.	DPJ	RJS	DLP
3. W/T	a. Verify that: (1) the outline(s) contain(s) the required number of control room and in-plant tasks, ✓ (2) no more than 30% of the test material is repeated from the last NRC examination, ✓ (3)* no tasks are duplicated from the applicants' audit test(s), and ✓ (4) no more than 80% of any operating test is taken directly from the licensee's exam banks. ✓	DPJ	RJS	DLP
	b. Verify that: (1) the tasks are distributed among the safety function groupings as specified in ES-301, ✓ (2) one task is conducted in a low-power or shutdown condition, ✓ (3) 40% of the tasks require the applicant to implement an alternate path procedure, ✓ (4) one in-plant task tests the applicant's response to an emergency or abnormal condition, and ✓ (5) the in-plant walk-through requires the applicant to enter the RCA. ✓	DPJ	RJS	DLP
	c. Verify that the required administrative topics are covered, with emphasis on performance-based activities.	DPJ	RJS	DLP
	d. Determine if there are enough different outlines to test the projected number and mix of applicants and ensure that no items are duplicated on successive days.	DPJ	RJS	DLP
4. GENERAL	a. Assess whether plant-specific priorities (including PRA and IPE insights) are covered in the appropriate exam section.	DPJ	RJS	DLP
	b. Assess whether the 10 CFR 55.41/43 and 55.45 sampling is appropriate.	DPJ	RJS	DLP
	c. Ensure that K/A importance ratings (except for plant-specific priorities) are at least 2.5.	DPJ	RJS	DLP
	d. Check for duplication and overlap among exam sections.	DPJ	RJS	DLP
	e. Check the entire exam for balance of coverage.	DPJ	RJS	DLP
	f. Assess whether the exam fits the appropriate job level (RO or SRO).	DPJ	RJS	DLP

	Printed Name / Signature	Date
a. Author	David P. Johnson / David P. Johnson	11-29-01
b. Facility Reviewer (*)	Robert J. Sochia / Robert J. Sochia	11-29-01
c. NRC Chief Examiner (#)	DAVID L PELTON / [signature]	12/31/01
d. NRC Supervisor	David C. Hills / David C. Hills	12/31/01

Note: * Not applicable for NRC-developed examinations.
\# Independent NRC reviewer initial items in Column "c;" chief examiner concurrence required.

Facility: Perry Date of Exam: 3/4/2002 Exam Level: RO

Tier	Group	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G *	Point Total
1. Emergency & Abnormal Plant Evolutions	1	2	1	2				5	2			1	13
	2	1	7	4				4	2			1	19
	3	0	1	1				1	1			0	4
	Tier Totals	3	9	7				10	5			2	36
2. Plant Systems	1	4	2	4	4	2	2	3	2	1	3	1	28
	2	4	0	0	4	2	1	1	3	1	2	1	19
	3	0	0	0	1	0	1	1	0	0	0	1	4
	Tier Totals	8	2	4	9	4	4	5	5	2	5	3	51

3. Generic Knowledge and Abilities	Cat 1	Cat 2	Cat 3	Cat 4	
	4	3	2	4	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. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by +/- 1 from that specified in the table based on NRC revisions. The final exam must total 100 points.
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.

/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295[illegible] Main Turbine Generator Trip / 3				X			AA1.07 Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: AC Electrical Distribution	3.3	1
2950[illegible] SCRAM / 1					X		AA2.06 Ability to determine and/or interpret the following as they apply to SCRAM: Cause of Reactor SCRAM	3.5	1
295[illegible] High Reactor Pressure / 3					X		AA2.01 Ability to determine and/or interpret the following as the apply to HIGH REACTOR PRESSURE: Reactor Pressure	4.1	1
2950[illegible] Low Reactor Water Level / 2				X			AA1.04 Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Reactor Water Cleanup	2.7	1
295[illegible] Low Reactor Water Level / 2	X						AK1.01 Knowledge of the operational implications of the following concepts as they apply to LOW REACTOR WATER LEVEL: Steam Carryunder	2.7	1
295[illegible] High Drywell Pressure / 5						X	2.4.1 Knowledge of EOP entry conditions and immediate action steps	4.3	1
295[illegible] Inadvertent Reactivity Addition / 1	X						AK1.03 Knowledge of the operational implications of the following concepts as they apply to INADVERTENT REACTIVITY ADDITION: Shutdown Margin	3.7	1
295[illegible] Incomplete SCRAM / 1				X			AA1.04 Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: Rod Control and Information System	3.4	1
295[illegible] High Drywell Pressure / 5		X					EK2.05 Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: RPS	3.9	1
2950[illegible] High Reactor Pressure / 3			X				EK3.09 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: Low-Low set initiation	3.7	1
2950[illegible] Reactor Low Water Level / 2			X				EK3.03 Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL: Spray Cooling	4.1	1
2950[illegible] SCRAM Condition Present and Power Abo[illegible] APRM Downscale or Unknown / 1				X			EA1.04 Ability to operate and /or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: SBLC	4.5	1
500[illegible] High Containment Hydrogen Conc. / 5				X			EA1.03 Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: Containment Atmosphere Control system	3.4	1
K/A Category Totals:	2	1	2	5	2	1	Group Point Total:		13

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2

?APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295? Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4			X				AK3.06 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Core Flow indication	2.9	1
295? Loss of Main Condenser Vacuum / 3			X				AK3.04 Knowledge of the reasons for the following responses as they apply to LOSS OF MAIN CONDENSER VACUUM: Bypass valve closure	3.4	1
295? Partial or Complete Loss of AC Pwr / 6				X			AA1.04 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF AC POWER: DC Electrical distribution systems	3.6	1
295? Partial or Complete Loss of DC Pwr / 6				X			AA1.01 Ability to operate and/or monitor the following responses as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: DC Electrical distribution systems	3.3	1
295? High Reactor Water Level / 2			X				AK3.08 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL: RCIC Steam supply valve closure	3.4	1
295? High CTMT Temperature / 5					X		AA2.03 Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (Mark III Containment Only): Containment Humidity	2.8	1
2950?? High Drywell Temperature / 5	X						AK1.01 Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Pressure/temperature relationship	3.3	1
2950?? High Suppression Pool Temp. / 5		X					AK2.01 Knowledge of the interrelations between HIGH SUPPRESSION POOL TEMPERATURE and the following: Suppression Pool Cooling	3.6	1
295?? Control Room Abandonment / 7						X	2.4.34 Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications	3.8	1
295? High Off-site Release Rate / 9		X					AK2.14 Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: PCIS/NSSSS	4.0	1
295?? Partial or Complete Loss of CCW / 8		X					AK2.01 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: System Loads	3.3	1
295?? Part. Or Comp. Loss of Inst. Air / 8			X				AK3.03 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Service Air isolations	3.2	1
295?? Inadvertent Cont. Isolation / 5 & 7		X					AK2.12 Knowledge of the interrelations between INADVERTENT CONTAINMENT ISOLATION and the following: Instrument Air/Nitrogen	3.1	1
2950?? Loss of CRD Pumps / 1				X			AA1.02 Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: RPS	3.6	1
2950?? Suppression Pool High Water Temp. /5				X			EA1.03 Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Temperature Monitoring	3.9	1

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

APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295... High CTMT Temperature / 5		X					EK2.01 Knowledge of the interrelations between HIGH CONTAINMENT TEMPERATURE (Mark III Containment Only) and the following: Containment Spray	3.2	1
2950.. High Drywell Temperature / 5									
2950.. High Suppression Pool Water Level / 5		X					EK2.07 Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following: Drywell/Containment Water Level	3.1	1
2950.. Low Suppression Pool Water Level / 5		X					EK2.08 Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: SRV Discharge submergence	3.5	1
2950.. High Sec. Cont. Area Rad. Levels / 9									
2950.. Sec. Cont. Ventilation High Rad. / 9									
2950.. High Off-site Release Rate / 9									
6000.. Plant Fire On Site / 8					X		AA2.17 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Systems that may be affected by the fire	3.1	1
K/A Category Point Totals:	1	7	4	4	2	1	Group Point Total:		19

Emergency and Abnormal Plant Evolutions - Tier 1/Group 3

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295[illegible] Loss of Shutdown Cooling / 4			X				AK3.01 Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: Raising reactor water level	3.3	1
295[illegible] Refueling Accidents / 8				X			AA1.03 Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Fuel Handling equipment	3.3	1
295[illegible] High Secondary Containment Area Temperature / 5		X					EK2.08 Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: Systems required for safe shutdown	3.8	1
295[illegible] Secondary Containment High Differential Pressure / 5							Suppressed		
295[illegible] 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
20[illegible] CR[illegible] Hydraulic System									X			A3.05 Ability to monitor automatic operations of the CONTROL ROD DRIVE HYDRAULIC SYSTEM including: Reactor water level	2.8	[illegible]
20[illegible] RMCS												Suppressed		
20[illegible] RCIS						X						K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the ROD CONTROL AND INFORMATION SYSTEM: IRM channel	3.0	[illegible]
20[illegible] Rec[illegible]culation Flow Control			X									K3.06 Knowledge of the effect that a loss or malfunction of the RECIRCULATION FLOW CONTROL SYSTEM will have on the following: Recirculation flow control valve position	3.7	[illegible]
20[illegible] RHR LPCI: Injection Mode										X		A4.06 Ability to manually operate and/or monitor in the control room: System reset following automatic initiation	3.9	[illegible]
206[illegible] HPCI												Suppressed		
207[illegible] Isolation (Emergency) Co[illegible]ser												Suppressed		
20[illegible] LPCS					X							K5.01 Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM: Indications of pump cavitation	2.6	[illegible]
209[illegible] HPCS			X									K3.01 Knowledge of the effect that a loss or malfunction of the HIGH PRESSURE CORE SPRAY SYSTEM (HPCS) will have on the following: Reactor Water level	3.9	1
209[illegible] HPCS		X										K2.03 Knowledge of electrical power supplies to the following: Initiation logic	2.8	1
211[illegible] SLC		X										K2.02 Knowledge of electrical power supplies to the following: Explosive valves	3.1	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp	Points
211[illegible] SLC											X	2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications	3.4	1
212[illegible] RPS							X					A1.11 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: System status lights and alarms	3.4	1
21[illegible] RP[illegible]					X							K5.02 Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements	3.3	[illegible]
215[illegible] IRM								X				A2.06 Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty Range switch	3.0	1
2150[illegible]3 IRM				X								K4.01 Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Rod withdrawal blocks	3.7	1
215[illegible] SRM			X									K3.04 Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on the following: Reactor power and indication	3.7	1
215[illegible] SRM						X						K6.02 Knowledge of the effect that a loss or malfunction of the following will have on the SOURCE RANGE MONITOR (SRM) SYSTEM: 24/48 volt DC power	3.1	[illegible]
215[illegible] APRM/LPRM				X								K4.06 Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Effects of detector aging on LPRM/APRM readings	2.6	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
216[illegible] Nuc[illegible] Boiler Instrumentation	X											K1.09 Knowledge of the physical connections and/or cause-effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following: Redundant reactivity control/ alternate rod insertion	3.7	1
217[illegible] RCIC								X				A2.12 Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve openings	3.0	1
217[illegible] RCIC										X		A4.04 Ability to manually operate and/or monitor in the control room: Manually initiated controls	3.6	1
218000 ADS				X								K4.03 Knowledge of AUTOMATIC DEPRESSURIZATION SYSTEM design feature(s) and/or interlocks which provide for the following: ADS logic control	3.8	1
223[illegible] Primary CTMT and Auxiliaries										X		A4.06 Ability to manually operate and/or monitor in the control room: Containment pressure (Mark III)	4.0	1
223[illegible] PCIS Nuclear Steam Supply Shut[illegible]	X											K1.15 Knowledge of the physical connections and/or cause-effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: High pressure core spray	3.4	1
239[illegible] SRV[illegible]				X								K4.09 Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Manual opening of the SRV	3.7	1
241[illegible] Reactor/Turbine Pressure Reg[illegible]			X									K3.30 Knowledge of the effect that a loss or malfunction of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM will have on the following: EGC	3.0	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
259[illegible] Reactor Feedwater							X					A1.02 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR FEEDWATER SYSTEM controls including: Feedwater inlet temperature	3.2	1
259[illegible] Reactor Water Level Control							X					A1.02 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including Reactor feedwater flow	3.6	1
261[illegible] SGTS	X											K1.07 Knowledge of the physical connections and/or cause-effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: Elevated stack release	3.1	1
264[illegible] EDGs	X											K1.06 Knowledge of the physical connections and/or cause-effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: Starting system	3.2	1
K/A Category Point Totals:	4	2	4	4	2	2	3	2	1	3	1	Group Point Total:		28

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
20[illegible] Control Rod and Drive Me[illegible]sm					X							K5.04 Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM: Rod sequence patterns	3.1	[illegible]
20[illegible] RSC[illegible]												Suppressed		
20[illegible] RWM												Suppressed		
202 Recirculation				X								K4.01 Knowledge of RECIRCULATION System design feature(s) and/or interlocks which provide for the following: 2/3 core coverage	3.9	1
202[illegible] Recirculation	X											K1.08 Knowledge of the physical connections and/or cause-effect relationships between RECIRCULATION SYSTEM and the following: AC Electrical	3.1	1
204 RWCU									X			A3.03 Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including: Response to system isolations	3.6	1
205 Shutdown Cooling System (RHR Shutdown Cooling mode)				X								K4.05 Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: Reactor cooldown rate	3.6	1
214 RPIS												Suppressed		
215 RBM												Suppressed		
219 RHR LPCI: Torus/Pool Cooling Mod[illegible]	X											K1.04 Knowledge of the physical connections and/or cause-effect relationships between RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE and the following: LPCI/RHR pumps	3.9	1
226 RHR LPCI: Containment Spray Sys[illegible] mode										X		A4.08 Ability to manually operate and/or monitor in the control room: System flow	3.2	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
23[illegible] RHR LPCI: Torus/Pool Spray Mo[illegible]												Suppressed		
239[illegible] Main and Reheat Steam										X		A4.09 Ability to manually operate and/or monitor in the control room: Reactor pressure	3.9	[illegible]
245[illegible] Main Turbine Gen And Aux[illegible]es								X				A2.07 Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of reactor/turbine pressure control systems	3.8	[illegible]
256[illegible] Reactor Condensate							X					A1.04 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CONDENSATE SYSTEM controls including: Hotwell level	2.9	1
262[illegible] AC Electrical Distribution											X	2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual	3.3	1
262[illegible] UPS (AC/DC)				X								K4.01 Knowledge of UNINTERRUPTIBLE POWER SUPPLY (AC/DC) design feature(s) and/or interlocks which provide for the following: Transfer from preferred power to alternate power supplies	3.1	1
263[illegible] DC Electrical Distribution				X								K4.02 Knowledge of DC ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Breaker interlocks, permissives, bypasses and cross ties	3.1	1
271[illegible] Offgas								X				A2.14 Ability to (a) predict the impacts of the following on the OFFGAS SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Offgas filter high differential pressure	2.6	1
272[illegible] Radiation Monitoring						X						K6.01 Knowledge of the effect that a loss or malfunction of the following will have on the RADIATION MONITORING SYSTEM: Reactor Protection System	3.0	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
286[illegible] Fire Protection					X							K5.02 Knowledge of the operational implications of the following concepts as they apply to FIRE PROTECTION SYSTEM: Effect of halon on fires	2.6	1
29[illegible] Secondary CTMT												Suppressed		
2900[illegible] Control Room HVAC	X											K1.03 Knowledge of the physical connections and/or cause-effect relationships between CONTROL ROOM HVAC and the following: Remote Air intakes	2.8	1
300[illegible] Instrument Air	X											K1.04 Knowledge of the connections and/or cause/effect relationships between INSTRUMENT AIR SYSTEM and the following: Cooling water to compressor	2.8	1
400[illegible] Component Cooling Water								X				A2.03 Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High/Low CCW temperature	2.9	1
K/A Category Point Totals:	4	0	0	4	2	1	1	3	1	2	1	Group Point Total:		19

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
215.. Traversing In-Core Probe												Suppressed		
233 Fuel Pool Cooling and Cleanup				X								K4.06 Knowledge of FUEL POOL COOLING AND CLEANUP design feature(s) and/or interlocks which provide for the following: Maintenance of adequate pool level	2.9	:
234 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	:
239 MSIV Leakage Control												Suppressed		
268 Radwaste														
288 Plant Ventilation											X	2.1.27 Knowledge of system purpose and/or function	2.8	1
290 Reactor Vessel Internals						X						K6.15 Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR VESSEL INTERNALS: ADS	3.1	1
K/A Category Point Totals:	0	0	0	1	0	1	1	0	0	0	1	Group Point Total:		4

Plant-Specific Priorities

System / Topic	Recommended Replacement for	Reason	Points
223.. A4.0? Primary Containment and Auxiliaries / Containment pressure	223001K6.12	Plant Specific high importance PSA for manually venting containment on high pressure	:
217.. A4.04 RCIC / Manually initiated controls	217000K5.03	Plant Specific high importance PSA requiring manual initiation of RCIC	:
Plant Specific Priority Total: (limit 10)			2

Category	K/A#	Topic	Imp.	Points
[illegible] Operations	2.1.11	Knowledge of less than one hour technical specification action statements for systems	3.0	1
	2.1.14	Knowledge of system status criteria which require the notification of plant personnel	2.5	1
	2.1.29	Knowledge of how to conduct and verify valve lineups	3.4	1
	2.1.30	Ability to locate and operate components / including local controls	3.9	1
	Total			4
[illegible]uipment Control	2.2.2	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels	4.0	1
	2.2.13	Knowledge of tagging and clearance procedures	3.6	1
	2.2.34	Knowledge of the process for determining the internal and external effects on core reactivity	2.8	1
	Total			3
[illegible] Control	2.3.1	Knowledge of 10CFR20 and related facility radiation control requirements	2.6	1
	2.3.9	Knowledge of the process for performing a containment purge	2.5	1
	Total			2
Em[illegible]ncy Procedures/Plan	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
	2.4.11	Knowledge of abnormal condition procedures	3.4	1
	2.4.12	Knowledge of general operating crew responsibilities during emergency operations	3.4	1
	2.4.19	Knowledge of EOP layout/symbols/ and icons	2.7	1
	Total			4
Tier 3 [illegible] Total (RO/SRO)				13

Tier/Group	Randomly Selected K/A	Reason for Rejection
1 / 1	500000EA1.05	No wetwell sprays at facility
1 / 2	295016AA1.03	RPIS not relevant to selected APE for facility
2 / 1	218000A1.02	Facility does not have ADS acoustic sensors
2 / 1	259002A2.08	Facility does not have FWCI (BWR-6)
2 / 1	203000K4.11	Facility does not have RHR Loop selection logic (BWR-6)
2 / 2	300000K3.03	Single unit facility does not have unit cross-ties for instrument air
1 / 1	295024EK2.01	Tier 1 Group 1 has more questions than available systems for RO level. EK2 for same EPE was previously selected. This topic was rejected to minimize chance for redundant questions. Additionally, this topic has a lower K/A value
2 / 2	239001K1.13	System interaction with suppressed system (MSIV Leakage control)
2 / 1	223001K6.12	Replaced by plant specific priority
2 / 1	217000K5.03	Replaced by plant specific priority
1 / 2	600000AA2.06	After further scrutiny, unable to write a viable exam question. It was determined this APE topic not applicable to Perry. The new topic (AA2.17) also has a higher Importance Rating.
2 / 1	261000K1.09	After further scrutiny, unable to write a viable exam question. It was determined this APE topic not applicable to Perry. There is no relationship between AEGTS and Primary Containment Isolation System at Perry.
2 / 2	263000K4.01	After further scrutiny, unable to write a viable exam question. It was determined this APE topic not applicable to Perry. The new topic (K4.02) also has the same Importance Rating.

Facility: Perry Date of Exam: 3/4/2002 Exam Level: SRO

Tier	Group	K/A Category Points											Point Total
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G *	
1. Emergency & Abnormal Plant Evolutions	1	3	7	2				8	4			2	26
	2	1	4	4				4	3			1	17
	Tier Totals	4	11	6				12	7			3	43
2. Plant Systems	1	2	2	3	2	2	2	1	2	0	4	3	23
	2	1	0	0	2	0	1	2	4	2	0	1	13
	3	0	0	0	1	1	0	1	0	0	1	0	4
	Tier Totals	3	2	3	5	3	3	4	6	2	5	4	40

3. Generic Knowledge and Abilities	Cat 1	Cat 2	Cat 3	Cat 4	
	4	5	2	6	17

Note:
1. Ensure that at least two topics from every K/A category are sampled within each tier (i.e., the "Tier Totals" in each K/A category shall not be less than two).
2. The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by +/- 1 from that specified in the table based on NRC revisions. The final exam must total 100 points.
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
295003 Partial or Complete Loss of AC Pwr / 6		X					**AK2.03 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF AC POWER and the following: A.C. electrical distribution system**	3.9	**1**
295003 Partial or Complete Loss of AC Pwr / 6				X			AA1.04 Ability to operate and/or monitor the following as they apply to PERTIAL OR COMPLETE LOSS OF AC POWER: DC Electrical distribution systems	3.7	1
295006 SCRAM / 1	X						AK1.03 Knowledge of the operational implications of the following concepts as they apply to SCRAM: Reactivity Control	4.0	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**
295007 High Reactor Pressure / 3					X		AA2.01 Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE: Reactor Pressure	4.1	1
295009 Low Reactor Water Level / 2				X			AA1.04 Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: Reactor Water Cleanup	2.7	1
295009 Low Reactor Water Level / 2	X						AK1.01 Knowledge of the operational implications of the following concepts as they apply to LOW REACTOR WATER LEVEL: Steam Carryunder	2.9	1
295010 High Drywell Pressure / 5					X		**AA2.01 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Leak rates**	3.8	**1**
295010 High Drywell Pressure / 5						X	2.4.1 Knowledge of EOP entry conditions and immediate action steps	4.6	1
295013 High Suppression Pool Temp. / 5				X			AA1.01 Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: Suppression Pool Cooling	3.9	1
295013 High Suppression Pool Temp. / 5		X					AK2.01 Knowledge of the interrelations between HIGH SUPPRESSION POOL TEMPERATURE and the following: Suppression Pool Cooling	3.7	1
295014 Inadvertent Reactivity Addition / 1	X						AK1.03 Knowledge of the operational implications of the following concepts as they apply to INADVERTENT REACTIVITY ADDITION: Shutdown Margin	4.0	1
295015 Incomplete SCRAM / 1				X			AA1.04 Ability to operate and/or monitor the following as they apply to INCOMPLETE SCRAM: Rod Control and Information System	3.7	1
295016 Control Room Abandonment / 7						X	2.4.34 Knowledge of RO tasks performed outside the main control room during emergency operations including system geography and system implications	3.6	1
295017 High Off-site Release Rate / 9		X					AK2.14 Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: PCIS/NSSSS	4.1	1
295023 Refueling Accidents Cooling Mode / 8				X			AA1.03 Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Fuel Handling equipment	3.6	1
295024 High Drywell Pressure / 5		X					EK2.05 Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following: RPS	4.0	1

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295024 High Drywell Pressure / 5					**X**		**EA2.01 Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: Drywell Pressure**	**4.4**	**1**
295025 High Reactor Pressure / 3			X				EK3.09 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: Low-Low set initiation	3.7	1
295026 Suppression Pool High Water Temp. /5				X			EA1.03 Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Temperature Monitoring	3.9	1
295027 High CTMT Temperature / 5		X					EK2.01 Knowledge of the interrelations between HIGH CONTAINMENT TEMPERATURE (Mark III Containment Only) and the following: Containment Spray	3.4	1
295030 Low Suppression Pool Water Level/5		**X**					**EK2.06 Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: Suppression Pool makeup**	**3.9**	**1**
295031 Reactor Low Water Level / 2			X				EK3.03 Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL: Spray Cooling	4.4	1
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1				X			EA1.04 Ability to operate and /or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: SBLC	4.5	1
295038 High Off-site Release Rate / 9		**X**					**EK2.10 Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Condenser Air Removal System**	**3.4**	**1**
500000 High Containment Hydrogen Conc. / 5				X			EA1.03 Ability to operate and monitor the following as they apply to HIGH CONTAINMENT HYDROGEN CONTROL: Containment Atmosphere Control System	3.2	1
K/A Category Totals:	3	7	2	8	4	2	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 & 4			X				AK3.06 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION: Core Flow indication	3.0	1
295002 Loss of Main Condenser Vacuum / 3			X				AK3.04 Knowledge of the reasons for the following responses as they apply to LOSS OF MAIN CONDENSER VACUUM: Bypass valve closure	3.6	1
295004 Partial or Complete Loss of DC Pwr / 6				X			AA1.01 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF DC POWER: DC Electrical distribution systems	3.4	1
295005 Main Turbine Generator Trip / 3				X			AA1.07 Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP: AC Electrical Distribution	3.3	1
295008 High Reactor Water Level / 2			X				AK3.08 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL: RCIC Steam supply valve closure	3.5	1
295011 High CTMT Temperature / 5					X		AA2.03 Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT TEMPERATURE (Mark III Containment Only): Containment Humidity	3.2	1
295012 High Drywell Temperature / 5	X						AK1.01 Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE: Pressure/temperature relationship	3.5	1
295018 Partial or Complete Loss of CCW / 8		X					AK2.01 Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: System Loads	3.4	1
295019 Part. Or Comp. Loss of Inst. Air / 8			X				AK3.03 Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Service Air isolations	3.2	1
295020 Inadvertent Cont. Isolation / 5 & 7		X					AK2.12 Knowledge of the interrelations between INADVERTENT CONTAINMENT ISOLATION and the following: Instrument Air/Nitrogen	3.2	1
295021 Loss of Shutdown Cooling / 4						X	**2.4.9 Knowledge of low power/shutdown implications in accident (e.g., LOCA or Loss of RHR) mitigation strategies**	**3.9**	**1**
295022 Loss of CRD Pumps / 1				X			AA1.02 Ability to operate and/or monitor the following as they apply to LOSS OF CRD PUMPS: RPS	3.6	1
295028 High Drywell Temperature / 5									
295029 High Suppression Pool Water Level / 5		X					EK2.07 Knowledge of the interrelations between HIGH SUPPRESSION POOL WATER LEVEL and the following: Drywell/Containment Water Level	3.2	1
295032 High Secondary Containment Area Temperature / 5		X					EK2.08 Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: Systems required for safe shutdown	3.9	1

E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295033 High Sec. Cont. Area Rad. Levels / 9									
295034 Sec. Cont. Ventilation High Rad. / 9				X			EA1.02 Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION: Process Radiation Monitoring System	4.0	1
295035 Secondary Containment High Differential Pressure / 5							Suppressed		
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		**AA2.15 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Requirements for establishing a fire watch**	**3.5**	**1**
K/A Category Point Totals:	1	4	4	4	3	1	Group Point Total:		17

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201005 RCIS						X						K6.04 Knowledge of the effect that a loss or malfunction of the following will have on the ROD CONTROL AND INFORMATION SYSTEM: IRM channel	3.2	1
202002 Recirculation Flow Control			X									K3.06 Knowledge of the effect that a loss or malfunction of the RECIRCULATION FLOW CONTROL SYSTEM will have on the following: Recirculation flow control valve position	3.7	1
203000 RHR/LPCI: Injection Mode										X		A4.06 Ability to manually operate and/or monitor in the control room: System reset following automatic initiation	3.9	1
206000 HPCI												Suppressed		
207000 Isolation (Emergency) Condenser												Suppressed		
209001 LPCS					X							K5.01 Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM: Indications of pump cavitation	2.7	1
209002 HPCS		X										K2.03 Knowledge of electrical power supplies to the following: Initiation logic	2.9	1
211000 SLC											**X**	**2.1.33: Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications**	**4.0**	**1**
212000 RPS					X							K5.02 Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM: Specific logic arrangements	3.4	1
215004 SRM			X									K3.04 Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on the following: Reactor power and indication	3.7	1
215004 SRM						X						K6.02 Knowledge of the effect that a loss or malfunction of the following will have on the SOURCE RANGE MONITOR (SRM) SYSTEM: 24/48 volt DC power	3.3	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
215005 APRM/LPRM				X								K4.06 Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Effects of detector aging on LPRM/APRM readings	2.8	1
216000 Nuclear Boiler Instrumentation	X											K1.09 Knowledge of the physical connections and/or cause-effect relationships between NUCLEAR BOILER INSTRUMENTATION and the following: Redundant reactivity control/ alternate rod insertion	4.0	1
217000 RCIC								**X**				**A2.12 Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve openings**	**3.0**	**1**
217000 RCIC										X		A4.04 Ability to manually operate and/or monitor in the control room: Manually initiated controls	3.6	1
218000 ADS											**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**
223001 Primary CTMT and Auxiliaries										X		A4.06 Ability to manually operate and/or monitor in the control room: Containment pressure (Mark III)	4.0	1
223002 PCIS/Nuclear Steam Supply Shutoff	X											K1.15 Knowledge of the physical connections and/or cause-effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the following: High pressure core spray	3.4	1
226001 RHR/LPCI: Containment Spray System mode										X		A4.08 Ability to manually operate and/or monitor in the control room: System flow	3.1	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
239002 SRVs				X								K4.09 Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Manual opening of the SRV	3.6	1
241000 Reactor/Turbine Pressure Regulator			X									K3.30 Knowledge of the effect that a loss or malfunction of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM will have on the following: EGC	3.0	1
259002 Reactor Water Level Control							X					A1.02 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Reactor feedwater flow	3.5	1
261000 SGTS		X										K2.03 Knowledge of electrical power supplies to the following: Initiation logic	2.5	1
262001 AC Electrical Distribution											X	2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual	3.3	1
264000 EDGs								X				**A2.03 Ability to (a) predict the impacts of the following on the EMERGENCY GENERATORS (Diesel/Jet) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Operating unloaded, lightly loaded, and highly loaded**	**3.4**	**1**
290001 Secondary CTMT												Suppressed		
K/A Category Point Totals:	2	2	3	2	2	2	1	2	0	4	3	Group Point Total:		23

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201001 CRD Hydraulic System									X			A3.05 Ability to monitor automatic operations of the CONTROL ROD DRIVE HYDRAULIC SYSTEM including: Reactor water level	2.8	1
201002 RMCS												Suppressed		
201004 RSCS												Suppressed		
201006 RWM												Suppressed		
202001 Recirculation				X								K4.01 Knowledge of RECIRCULATION System design feature(s) and/or interlocks which provide for the following: 2/3 core coverage	3.9	1
204000 RWCU									X			A3.03 Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including: Response to system isolations	3.6	1
205000 Shutdown Cooling System (RHR Shutdown Cooling mode)								X				**A2.02 Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low shutdown cooling suction pressure**	**2.7**	**1**
214000 RPIS												Suppressed		
215002 RBM												Suppressed		

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
215003 IRM								X				A2.06 Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty Range switch	3.2	1
219000 RHR/LPCI: Torus/Pool Cooling Mode	X											K1.04 Knowledge of the physical connections and/or cause-effect relationships between RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE and the following: LPCI/RHR pumps	3.9	1
230000 RHR/LPCI: Torus/Pool Spray Mode												Suppressed		
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	3.4	1
239003 MSIV Leakage Control												Suppressed		
245000 Main Turbine Gen. And Auxiliaries								X				A2.07 Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of reactor/turbine pressure control systems	3.9	1
259001 Reactor Feedwater							X					A1.02 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR FEEDWATER SYSTEM controls including: Feedwater inlet temperature	3.3	1
262002 UPS (AC/DC)				X								K4.01 Knowledge of UNINTERRUPTIBLE POWER SUPPLY (AC/DC) design feature(s) and/or interlocks which provide for the following: Transfer from preferred power to alternate power supplies	3.4	1

System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
263000 DC Electrical Distribution											**X**	**2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm**	**3.6**	**1**
271000 Offgas								X				**A2.02 Ability to (a) predict the impacts of the following on the OFFGAS SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low dilution steam flow**	**3.1**	**1**
272000 Radiation Monitoring						X						K6.01 Knowledge of the effect that a loss or malfunction of the following will have on the RADIATION MONITORING SYSTEM: Reactor Protection System	3.2	1
286000 Fire Protection														
290003 Control Room HVAC														
300000 Instrument Air														
400000 Component Cooling Water														
K/A Category Point Totals:	1	0	0	2	0	1	2	4	2	0	1	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							K5.04 Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD AND DRIVE MECHANISM: Rod sequence patterns	3.4	1
215001 Traversing In-Core Probe												Suppressed		
233000 Fuel Pool Cooling and Cleanup				X								K4.06 Knowledge of FUEL POOL COOLING AND CLEANUP design feature(s) and/or interlocks which provide for the following: Maintenance of adequate pool level	3.2	1
239001 Main and Reheat Steam										X		A4.09 Ability to manually operate and/or monitor in the control room: Reactor pressure	3.9	1
256000 Reactor Condensate							X					A1.04 Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CONDENSATE SYSTEM controls including: Hotwell level	2.9	1
268000 Radwaste														
288000 Plant Ventilation														
290002 Rector Vessel Internals														
K/A Category Point Totals:	0	0	0	1	1	0	1	0	0	1	0	Group Point Total:		4

Plant-Specific Priorities

System / Topic	Recommended Replacement for	Reason	Points
223001A4.06 Primary Containment and Auxiliaries / Containment pressure	223001K6.12	Plant Specific high importance PSA for manually venting containment on high pressure	1
217000A4.04 RCIC / Manually initiated controls	217000K5.03	Plant Specific high importance PSA requiring manual initiation of RCIC	1
Plant-Specific Priority Total: (limit 10)			2

Category	K/A#	Topic	Imp.	Points
Conduct of Operations	**2.1.11**	**Knowledge of less than one hour technical specification action statements for systems**	**3.8**	**1**
	2.1.14	**Knowledge of system status criteria which require the notification of plant personnel**	**3.3**	**1**
	2.1.29	Knowledge of how to conduct and verify valve lineups	3.3	1
	2.1.30	Ability to locate and operate components / including local controls	3.4	1
	Total			4
Equipment Control	2.2.2	Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels	3.5	1
	2.2.12	**Knowledge of surveillance procedures**	**3.4**	**1**
	2.2.13	Knowledge of tagging and clearance procedures	3.8	1
	2.2.29	**Knowledge of SRO fuel handling responsibilities**	**3.8**	**1**
	2.2.34	Knowledge of the process for determining the internal and external effects on core reactivity	3.2	1
	Total			5
Radiation Control	**2.3.1**	**Knowledge of 10CFR20 and related facility radiation control requirements**	**3.0**	**1**
	2.3.9	**Knowledge of the process for performing a containment purge**	**3.4**	**1**
	Total			2

Emergency Procedures/Plan	**2.4.39**	**Knowledge of the RO's responsibilities in Emergency Plan implementation**	**3.1**	**1**
	2.4.11	**Knowledge of abnormal condition procedures**	**3.6**	**1**
	2.4.3	**Ability to identify post accident instrumentation**	**3.8**	**1**
	2.4.41	**Knowledge of the emergency action level thresholds and classifications**	**4.1**	**1**
	2.4.12	Knowledge of general operating crew responsibilities during emergency operations	3.9	1
	2.4.19	Knowledge of EOP layout/ symbols/ and icons	3.7	1
	Total			6
Tier 3 Point Total (RO/SRO)				17

Tier/Group	Randomly Selected K/A	Reason for Rejection
1 / 1	500000EA1.05	No wetwell sprays at facility
1 / 1	295016AA1.03	RPIS not relevant to selected APE for facility
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2 / 1	203000K4.11	Facility does not have RHR Loop selection logic (BWR-6)
2 / 2	300000K3.03	Single unit facility does not have unit cross-ties for instrument air
2 / 3	239001K1.13	System interaction with suppressed system (MSIV Leakage control)
1 / 1	295037EA1.07	EPE topic area interfacing with suppressed system (RMCS)
1 / 1	295014AK1.04	Facility does not use PCIOMR
2 / 1	223001K6.12	Replaced by plant specific priority
2 / 1	217000K5.03	Replaced by plant specific priority
1 / 1	295007AK2.04	This APE should have been a 'common' question (RO and SRO). This change will reduce the total number of exam questions by one. The new topic (AA2.01) also has a higher Importance Rating.
1 / 2	600000AA2.06	After further scrutiny, unable to write a viable exam question. It was determined this APE topic not applicable to Perry. The new topic (AA2.15) also has a higher Importance Rating.
2 / 2	271000A2.14	After further scrutiny, unable to write a viable exam question. The new topic (A2.02) also has a higher Importance Rating.

Facility: Perry
Examination Level: RO
Date of Examination: 3/4/2002
Operating Test Number: 1

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Shift Turnover	2.1.3 (3.0) – Knowledge of Shift Turnover Requirements JPM: Complete a Shift Relief/Turnover checklist as the oncoming operator
	Jet Pump Operability	2.1.7 (3.7) – Ability to Evaluate Plant Performance and Make Operational Judgements Based on Operating Characteristics / Reactor Behavior / Instrument Interpretation JPM: Determine Jet Pump operability
A.2	Tagging	2.2.13 (3.6) – Knowledge of Tagging and Clearance Procedures JPM: Establish equipment isolation boundaries
A.3	Radiation Control Requirements	2.3.1 (2.6) – Knowledge of 10CFR20 and Related Facility Radiation Control Requirements JPM: Evaluate requirements of RWP and perform appropriate actions for personnel contamination
A.4	Personnel Accountability	2.4.39 (3.3) - Knowledge of RO's Responsibilities in Emergency Plan Implementation JPM: Perform Site Accountability actions from outside the Control Room.

Facility: Perry
Examination Level: SRO

Date of Examination: 3/4/2002
Operating Test Number: 1

Administrative Topic/Subject Description		Describe method of evaluation: 1. ONE Administrative JPM, OR 2. TWO Administrative Questions
A.1	Shift Turnover	2.1.3 (3.4) – Knowledge of Shift Turnover Requirements JPM: Complete a Shift Relief/Turnover Checklist as the oncoming operator
	Feedwater Temperature Reduction Ops	2.1.7 (4.4) – Ability to Evaluate Plant Performance and Make Operational Judgements Based on Operating Characteristics / Reactor Behavior / Instrument Interpretation JPM: Prepare for Feedwater Temperature Reduction Operations
A.2	Risk Assessment	2.2.17 (3.5) – Knowledge of the Process for Managing Maintenance Activities During Power Operations JPM: Perform an On-Line Risk Determination
A.3	Radiation Control requirements	2.3.1 (3.0) – Knowledge of 10CFR20 and Related Facility Radiation Control Requirements JPM: Evaluate requirements of RWP and perform appropriate actions for personnel contamination
A.4	Emergency Plan	2.4.29 (4.0) – Knowledge of the Emergency Plan JPM: Classify an Emergency event, make Protective Action Recommendations, and complete paperwork for notification of Off-Site authorities

Facility: Perry **Date of Examination**: 3/4/2002

Exam Level RO/SROI **Operating Test No.**: 1

B.1: Control Room Systems

	System	JPM Description	Type Code*	Safety Function
S1	RRS 202001	Shift Recirculation Pump B from Slow Speed to Fast Speed and Raise Reactor Power using Recirculation Flow	SNA	1
S2	CRDH 201001/295031	CRD Alternate Injection for Level Control	MSA	2
S3	Control Room HVAC 290003	Shift CR HVAC & Emergency Recirculation from Emergency to Normal	SD	9
S4	RCIC 217000	RCIC Startup from Standby Readiness (CST to CST)	MASL	4
S5	RHR/LPCI 226001	Terminate Containment Spray RHR Loop A	NS	5
S6	DG 264000	Remotely Transfer Bus EH12 to the Alternate Preferred Source from the DG	SN	6
S7	MRSS 239001	Opening Inboard Main Steam Line Drain Valve	SN	3

B.2: Facility Walk-Through

	System	JPM Description	Type Code*	Safety Function
P1	SLCS 211000	Commence Alternate Boron Injection	DR	1
P2	RHR 203000/295031	Perform RHR Loop B Alternate Injection	DR	2
P3	Fire Protection 286000	Initiate CR Subfloor CO2 from Outside CR	MA	8

* **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: Perry **Date of Examination**: 3/4/2002

Exam Level: SROU **Operating Test No.:** 1

B.1: Control Room Systems

	System	JPM Description	Type Code*	Safety Function
S1				
S2				
S3				
S4	RCIC 217000	RCIC Startup from Standby Readiness (CST to CST)	MASL	4
S5				
S6	DG 264000	Remotely Transfer Bus EH12 to the Alternate Preferred Source from the DG	SN	6
S7	MRSS 239001	Opening Inboard Main Steam Line Drain Valve	SN	3

B.2: Facility Walk-Through

	System	JPM Description	Type Code*	Safety Function
P1				
P2	RHR 203000/295031	Perform RHR Loop B Alternate Injection	DR	2
P3	Fire Protection 286000	Initiate CR Subfloor CO2 from Outside CR	MA	8

* **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: Perry Scenario No.: 1 Op-Test No.: 2002-01

Examiners: ______________________ Operators: ______________________

Initial Conditions: A Reactor startup is in progress following a brief outage. Reactor power is being held at 70% power per SCC request. A xenon transient is in progress. RHR B is in secured status for preventive maintenance on the pump breaker. RHR B was declared inoperable five hours ago per Tech. Spec.3.5.1, Action A; 3.6.1.7, Action A; and 3.6.2.3, Action A. The MFP is in secured status to support recirc valve actuator work. SRV F041E is weeping and causing the Suppression Pool to slowly heatup. The OPRMs are functional but are inoperable per Tech. Spec. 3.3.1.3. Required Action A.3 has been implemented.

Turnover: 1. BOP operator place RHR A in suppression pool cooling mode and lower Suppression Pool temperature to 75 F. ESW A and ECC A are in operation. 2. Maintain 70% power.

Target Critical Tasks: Emergency Depressurization, Restore RPV water level

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP)	Startup RHR A in Suppression Pool cooling mode
2	MV06: 1E12-F064A	C (BOP)	RHR Min Flow valve (F064A) fails due to mechanical binding after RHR flow is established (TS 3.5.1.C, 3.6.1.7.B, 3.6.2.3.B)
3	CN02: 1C11R0600 100%	I (BOP)	CRDH flow controller failure in Auto mode
	RD12R1447	C (RO)	Single control rod drift inward (14-47)
4	Various	C (RO)	Reactor Feed Pump A bearing failure
		R (RO)	Lower reactor power to 63% using recirc flow
		N (RO)	Remove RFPT A from service
	ZL1N27R0425A	I (RO)	RFP A manual speed control dial pot failure (as is)
5	FW02 - 50%	M (All)	Feedwater System Pipe Break inside Drywell / Reactor Scram
	CP01: 1E22C0001	C (BOP)	HPCS Pump shaft breaks
	MV04: 1E51F0013	C (BOP)	RCIC Injection Valve (F013) failure to Auto open
	RC03	C (BOP)	RCIC Turbine mechanical trip latch failure
6	TH02C 75%	C (All)	Recirc Bottom Head Drain pipe break (5 minute ramp)
	ZD1B21S34B	C (BOP)	ADS B Inhibit Switch failure in Normal position
7		M	RPV emergency depressurization / Inject with low pressure ECCS to maintain adequate core cooling

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

Objectives:

The BOP operator places RHR A in Suppression Pool Cooling mode to lower Suppression Pool temperature to 75 F. When RHR is initiated, the RHR minimum flow valve will remain open after flow is established. This will make RHR inoperable and the Tech. Specs. will be consulted.

Immediately after Tech Specs are referenced, the CRDH Flow Controller will fail in the Auto mode causing a single control rod to drift inward. The crew will recognize the drifting control rod and enter ONI-C51, Unplanned Change in Reactor Power or Reactivity. When the crew identifies the failed CRDH Flow Controller, the controller will be placed in Manual and CRDH parameters will be returned to normal allowing the drifted control rod to settle at notch 00.

After a plan is implemented to recover the drifted control rod, RFP A will experience sustained high bearing vibration requiring reactor power to be lowered to 63% using recirc flow to allow the RFP to be removed from service. After the power reduction, as RFP A is being removed from service, the manual speed control dial potentiometer will fail requiring a trip of the RFP to remove it from service. When RFP A is tripped, a feedwater pipe breaks in the Drywell resulting in a reactor scram.

Following the scram, the HPCS pump will be unavailable due to a shaft break. When RCIC is initiated for level control, the RCIC injection valve will fail to open automatically requiring operator action to manually open the injection valve. After injection is established with RCIC, the RCIC Turbine mechanical trip latch will fail making RCIC unavailable for injection,

A small Recirc pipe break in the Drywell will develop and slowly increase in severity resulting in rising drywell temperature and pressure and lowering RPV water level. As reactor level continues to lower, the crew will place alternate injection systems in service, emergency depressurize the RPV, and maintain adequate core cooling with low pressure ECCS systems in accordance with PEI-B13, RPV Control (Non-ATWS). However, when ADS is inhibited, the ADS Inhibit Switch B will fail which may require the crew to take action to prevent an unintended ADS blowdown.

Discussion of Safety Significance for scenario 1

The BOP operator will place RHR in Suppression Pool Cooling. When RHR flow is initiated, the RHR Min flow valve stays open after flow is established. The BOP operator must note that the min flow valve has failed to close and inform the SRO because with the minimum flow valve failed open, RHR A is inoperable.

The CRDH Flow Controller will fail open in the Auto mode causing a single control rod to drift inward. The RO must recognize that a control rod is drifting and determine which rod is drifting. The drifting control rod is safety significant because it directly effects core reactivity and core power distribution. Operators will be required to determine the cause of the drifting control rod (failed CRDH flow controller) and begin actions to recover the control rod.

After a plan is implemented to recover the drifted control rod, RFP A will experience sustained high bearing vibration. This is safety significant because it will require the operators to lower reactor power in a controlled manner and remove the RFP from service. Removing the RFP from service is safety significant because the operation directly affects RPV water level control. As the RFP is being removed from service, the manual speed control dial pot will fail 'as is' requiring a trip of the RFP, directly leading to a Feedwater System pipe break in the Drywell and a reactor scram.

Following the scram, the HPCS pump will be unavailable due to a shaft break. When RCIC is initiated for level control, the RCIC injection valve will fail to open automatically. The RCIC injection valve failure is safety significant because RCIC is the only normal high-pressure injection system available. Operator action will be required to manually open the injection valve allowing RCIC injection. After manual flow control is established with RCIC, the RCIC Turbine mechanical trip latch will fail making RCIC unavailable for injection.

A small Recirc pipe break in the Drywell will develop and slowly increase in severity resulting in rising Drywell temperature and pressure and lowering RPV water level. The Recirc pipe break is safety significant because it directly contributes to the loss of coolant inventory requiring the crew to place alternate injection systems in service and prepare for RPV emergency depressurization to allow injection with low pressure ECCS systems. As RPV level continues to lower and ADS is inhibited, ADS Inhibit Switch B will fail. Failure of the capability to inhibit ADS is safety significant because it places the crew in a condition not anticipated by PEI-B13, RPV Control (Non-ATWS), requiring the crew to take action to prevent unintended RPV emergency depressurization.

Facility: Perry Scenario No.: 2 Op-Test No.: 2002-01

Examiners: ______________________ Operators: ______________________

Initial Conditions: The plant is operating at 100% power. A xenon transient is in progress. RHR B is in secured status for preventive maintenance on the pump breaker. RHR B was declared inoperable five hours ago per Tech. Spec.3.5.1, Action A; 3.6.1.7, Action A; and 3.6.2.3, Action A. The OPRMs are functional but are inoperable per Tech. Spec. 3.3.1.3. Required Action A.3 has been implemented. HPCS testing is scheduled to support flow rate testing.

Turnover: 1. Place HPCS in full flow test mode to the suppression pool. HPCS ESW and HPCS Pump Room Cooler are in operation. 2. Maintain 100% power.

Target Critical Tasks: Initiate action to shutdown the reactor, Inhibit ADS, Terminate and Prevent injection into the RPV, Emergency Depressurization, Restore RPV water level.

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP)	Start HPCS in full flow test mode to the Suppression Pool
2	CP03: 1E22C0001 100%	C (BOP)	HPCS Pump flow degradation (2 minute ramp) (TS 3.5.1. B and C)
3	AD01N	C (BOP) R (RO)	ADS/SRV B21-F047H cycling (TS 3.5.1.E, F and H / TS 3.0.3) Lower reactor power to 90% using recirc flow
4	CN03: 1C34R060B 20%	I (RO)	Reactor Feed Pump Controller B oscillations (1 minute ramp)
5	ED06I TC05 10%	C (BOP) C (ALL)	Loss of 480Vac Bus F-1-E Turbine Control EHC leak / Main Turbine trip and reactor scram
6	RD15 SL01A SL01B	C (RO) M (All) C (BOP)	Failure of RPS and ARI to automatically shutdown the reactor ATWS SLC Squib Valve failures, C41-F004A and C41-F004B
7	CB01: 1N27C0001A CB01: 1N27C0001B CB01: 1N27C0001C CB01: 1N27C0001D	C (RO) M (All)	All RFBPs trip Loss of all Feedwater capability
8	 RV04: 1B21-F0041F	M (All) C (BOP)	RPV emergency depressurization / Inject with low pressure ECCS to maintain adequate core cooling ADS/SRV B21- F041F failure closed

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

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Scenario Objectives
Safety Significance Discussions
Scenario 2

Objectives:

The BOP operator will place HPCS in the full flow test mode to support flowrate testing. After HPCS flow is stabilized for the test, HPCS Pump flow will be degraded requiring the HPCS System to be declared inoperable.

After Tech. Specs. have been consulted, an ADS/SRV will open and cycle due to shorted switch contacts; the crew will enter ONI-B21-1, SRV Inadvertent Opening/Stuck Open, evacuate the Containment, reduce reactor power to 90% using recirc flow, and de-energize the SRV solenoids by removing control power fuses. After the SRV closes, Tech Specs must again be consulted and the loss of HPCS and ADS will require a Tech. Spec. 3.0.3 entry.

The crew will respond to a RPV level transient due to Feedwater Flow Controller A oscillations that require entry into ONI-C34, Feedwater Flow Control Malfunction, and manual control of feedwater flow.

After conditions have stabilized, a loss of 480 Volt Bus F-1-E will result in a loss of the running EHC Pump B, TBCC Pump C, and CVCW Chiller A (in addition to other loads). Following auto start of standby EHC Pump A, the EHC System will develop a leak which will slowly increase in severity until the reactor is manually scrammed and the Main Turbine is tripped or the Main Turbine trips automatically.

When the reactor scrams, the control rods will fail to fully insert due to blockage in the scram discharge volume. PEI-B13 (RPV Control-ATWS) is entered and executed to stabilize the plant. SLC squib valves will fail to fire.

After PEI ATWS actions are underway and RPV level has reached L2, (130" above TAF), Feedwater System capability is lost and the RCIC system will be unable to provide adequate makeup. Therefore, the crew will emergency depressurize the RPV to allow for low pressure ECCS injection. Two ADS/SRVs will fail to operate during the emergency depressurization.

Discussion of Safety Significance for scenario 2

The BOP operator will place HPCS in the full flow test mode. After HPCS flow is stabilized for the test, the BOP operator must note that HPCS Pump flow is degraded. This is safety significant because the HPCS System is inoperable and unavailable for core cooling.

Next, an ADS/SRV will open and cycle due to shorted switch contacts; the crew will enter ONI-B21-1, evacuate the Containment, reduce reactor power, and deenergize the SRV solenoids by removing control power fuses. This is safety significant because the cycling SRV directly affects core reactivity and results in a rising Suppression Pool temperature. When SRV fuses are pulled, the crew must determine that the ADS/SRV is inoperable and unavailable. This is safety significant because it further degrades the status of the high-pressure ECCS systems, thus requiring entry into Tech. Spec. 3.0.3.

The crew will then respond to a RPV level transient due to Feedwater Flow Controller A oscillations. This is significant because it requires manual control of RPV water level to terminate the level transient.

After conditions have stabilized, a loss of 480 Volt Bus F-1-E will result in a loss of running EHC Pump B, TBCC Pump C, and CVCW Chiller A (in addition to other loads). This is safety significant because it requires the crew to evaluate plant status and enter the appropriate ONIs for the 480-Volt bus failure and TBCC Pump trip. Following the auto start of the standby EHC Pump A, a turbine control EHC leak will develop. This is safety significant because the crew must recognize the EHC leak, manually scram the reactor, and trip the Main Turbine before the Main Turbine trips automatically.

When the reactor is scrammed, the control rods fail to fully insert due to blockage in the scram discharge volume. The failure of control rods to insert is safety significant because it will require the operators to take PEI ATWS actions to shutdown the reactor, control reactor level, and control reactor pressure.

During ATWS actions, the SLC squib valves will fail to fire. This failure is safety significant because it will lengthen the time required to achieve reactor shutdown.

Following the loss of the Feedwater System, the crew must determine that the RCIC System will be unable to provide adequate makeup, thereby challenging the ability to maintain adequate core cooling. This is safety significant because emergency depressurization will be required to establish controlled injection with low pressure ECCS to assure adequate core cooling. Two ADS/SRVs will fail to operate during the emergency depressurization. Failure of the ADS SRVs to open is safety significant because the crew must recognize the failure and open additional SRVs.

Facility: Perry Scenario No.: 3 Op-Test No.: 2002-01

Examiners: ______________________ Operators: ______________________

Initial Conditions: Reactor startup is in progress with the plant at 5% of rated power. RHR B is in secured status for preventive maintenance on the pump breaker. The OPRMs are functional but are inoperable per Tech. Spec. 3.3.1.3. Required Action A.3 has been implemented.

Turnover: Plant startup continues: withdraw control rods to 10% power, transfer the Reactor Mode Switch to RUN, and continue power ascension. All required MODE 1 change paperwork has been reviewed and approved.

Target Critical Tasks: Emergency Depressurization, RPV Flooding to restore and maintain adequate core cooling

Event No.	Malf. No.	Event Type*	Event Description
1		R (RO)	Increase reactor power to 10% using control rods
	RD01:R1043 8%	C (RO)	Control rod 10-43 stuck at position 8
2	NM02H 100%	I (RO)	IRM H failure upscale (bypass failed IRM) (TS 3.3.1.1 and OR 6.2.3)
3		N (RO)	Verify NI overlap / Transfer Reactor Mode Switch to RUN / Withdraw IRMs
4	RD17A 50%	C (BOP)	CRDH Pump A trip due to loss of lube oil.
		N (BOP)	Perform CRD Pump trip recovery
	RD05R5443		Accumulator fault HCU 54-43 (TS 3.1.5) (1 minute time delay)
5	CP02: OP41C001B	C (BOP)	Service Water Pump 'B' trip due to shaft seizure (start standby Service Water Pump)
6	bat or/seismic_2	C (All)	Seismic Event (OBE)
	RP01A	C (All)	RPS 'A' EPA Breaker Trip (loss of RPS Bus 'A') (30 second time delay)
	TH02A / TH02B 100%	M (All)	Recirc Loop pipe rupture (reactor scram on high Drywell pressure) (TH02A - 6 minute time delay & 5 minute ramp) (TH02B - 8 minute time delay & 5 minute ramp)
	MV08: OP43F0215	C (BOP)	NCC Drywell Isolation Valve P43-F215 failure when valve becomes fully closed
	RD01R4219	C (RO)	Control rod 42-19 stuck at position 12 (during scram)
7	bat ms/losslevel2	I (All)	Loss of all RPV level indication
		M (All)	Emergency Depressurization / RPV Flooding to restore and maintain adequate core cooling

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

Objectives:

The crew will continue the startup. Prior to placing the Reactor Mode Switch in RUN, control rod 10-43 will not withdraw using normal drive water pressure. This will require the crew to take action per ONI-C11, Inability to Move Control Rods, and SOI-C11 (RCIS) to free the stuck control rod.

After the control rod is moving normally, IRM H will fail upscale resulting in a RPS half-scram. The startup will be placed on hold while Tech Specs are referenced, the IRM is bypassed, and the half-scram is reset.

The startup continues and after the Reactor Mode Switch has been placed in RUN, the running CRDH Pump will trip due to a loss of lube oil, requiring the standby CRDH Pump to be started. While performing CRD pump trip recovery, an HCU accumulator fault alarm will be received requiring the crew to monitor for additional accumulator fault alarms.

After the standby CRDH Pump is started, Service Water Pump B will trip. ONI-P41, Loss of Service Water, will be entered requiring manual start of a standby Service Water Pump.

Immediately after a standby Service Water Pump is started, a seismic event occurs which causes an RPS EPA breaker to trip and a Recirc Loop pipe break in the Drywell, resulting in rising Drywell pressure and temperature. NCC Drywell Isolation Valve P43-F215 will fail when it is fully closed further degrading the crew's ability to control Drywell temperature.

The reactor will be manually scrammed or will automatically scram; however, one control rod will fail to insert.

Following the scram, rising Drywell temperature will result in a loss of all level indication. RPV Flooding and Emergency Depressurization is performed and low-pressure injection systems are used to maintain adequate core cooling.

Discussion of Safety Significance for scenario 3:

As the startup continues, prior to placing the Reactor Mode Switch in RUN, one control rod will not withdraw using normal drive water pressure requiring the crew to take action to get the control rod to move. This is safety significant because the actions directly affect core reactivity.

After the control rod is unstuck, an IRM will fail upscale. This is safety significant because it will result in a RPS half-scram, require the crew to determine the IRM is inoperable, bypass the IRM, and reset the half-scram.

As the startup continues and the Reactor Mode Switch has been placed in RUN, the running CRDH Pump will trip that requires the standby CRDH Pump to be started. An accumulator fault alarm will also occur. This is safety significant because Tech. Specs. would require a manual reactor scram after 20 minutes if an additional HCU accumulator fault alarm were to be received.

After the standby CRDH Pump is started, a Service Water Pump will trip. This will require a manual start of a standby Service Water Pump to avoid high temperatures on components and systems cooled by the Service Water System.

After a standby Service Water Pump has been started, a seismic event occurs which causes a RPS EPA breaker to trip and a Recirc pipe break in the Drywell. This is safety significant because the reactor will be manually scrammed or will automatically scram following the seismic event. One control rod will fail to insert following the scram. The failure of the control rod to insert is safety significant because it will require the crew to make a timely assessment of control rod positions to determine if ATWS actions are necessary.

Following the scram, conditions in the Drywell will cause a loss of all level indication. This is safety significant because the crew must enter RPV Flooding and emergency depressurize the RPV in order to allow low-pressure injection systems to be used to maintain adequate core cooling.

Facility: Perry Scenario No.: 4 Op-Test No.: 2002-01

Examiners: ______________ Operators: ______________

Initial Conditions: The plant is operating at 100% power. RHR B is in secured status for preventive maintenance on the pump breaker. RHR B was declared inoperable five hours ago per Tech. Spec.3.5.1, Action A; 3.6.1.7, Action A; and 3.6.2.3, Action A. The OPRMs are functional but are inoperable per Tech. Spec. 3.3.1.3. Required Action A.3 has been implemented.

Turnover: 1. Shift NCC Pumps (start NCC Pump C and shutdown NCC Pump A).

Target Critical Tasks: Manually start RHR Pump A (failure to auto start), Emergency Depressurization

Event No.	Malf. No.	Event Type*	Event Description
1		N (BOP)	Shift NCC Pumps (start NCC Pump C and shutdown NCC Pump A)
2	CN02: 1P44R0450 0%	I (BOP)	RFPT A Lube Oil Temp controller failure in Auto mode
3	PT02: 1C34N0004A 17%	I (RO)	Reactor Narrow Range Level Transmitter N004A Offset (3 minute ramp) (ORM 6.2.1.3)
4	RF FW66 TH12B	C (RO) C (RO)	RFPT A spurious trip Reactor Recirculation FCV B runback failure (TS 3.4.1)
5	CP02: 1P44C0001A SW03 25%	C (BOP) C (ALL) R (RO)	TBCC Pump A trip (start standby TBCC Pump) TBCC System Process Piping Leakage (1 minute time delay and 3 minute ramp) Fast reactor shutdown required Decrease reactor power to 66% using recirc flow (58 Mlbs/hr)
6	TH28 1% PC01A 0% CB04: 1E12C0002A	M (All) C (BOP)	MSL Break in Drywell DW/CNTMT Bypass Leakage (to be modified in Event #7) RHR Pump A fails to auto start on Drywell high pressure (required for Containment Spray mode)
7	CB01: 1E12C0002A PT01: 1B21N0081C 5%	C (BOP) M (All) I (RO)	RHR Pump A trips when flow is aligned to containment spray RPV emergency depressurization to control Containment pressure Reactor Level Transmitter N081C failure downscale

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

Objectives:

With the plant operating at 100% power, the BOP operator will shift NCC Pumps.

After NCC Pump C has been placed in service and NCC pump A has been shutdown, RFPT A Lube Oil Temp controller will fail closed in Auto requiring the crew to place the controller in Manual and restore lube oil temperature to normal.

After lube oil temperature is restored, the Narrow Range level transmitter for the in-service channel fails low (offset). The crew enters ONI-C34, Feedwater Flow Control Malfunction, to select an operable Narrow Range level channel and restore level to normal.

When conditions have stabilized, RFPT A will trip. During the transient, Reactor Recirculation FCV B will fail to automatically runback requiring Technical Specifications to be referenced for a loop flow mismatch.

After level and power are stabilized, TBCC Pump A will trip. ONI-P44, Loss of TBCC, will be entered requiring manual start of the standby TBCC Pump.

Immediately following start of the standby TBCC Pump, a TBCC System piping leak will be initiated and grow progressively worse until a fast reactor shutdown is required.

Following the scram, a MSL pipe break in the Drywell occurs resulting in an MSIV isolation and rising Containment pressure.

When signaled to start on high Drywell pressure, RHR Pump A will not automatically start and must be manually started. When aligned for containment spray, RHR Pump A breaker will trip. Eventually the RPV must be depressurized to control the Containment pressure rise. During emergency depressurization, Reactor Level Transmitter N081C will fail downscale.

Discussion of Safety Significance for scenario 4

After NCC Pump C has been placed in service and NCC Pump A has been shutdown, RFPT A Lube Oil Temp controller will fail closed. This is safety significant because failure to recognize and correct the failure would eventually result in RFP bearing damage and a loss of the RFP, thereby challenging RPV water level control.

After lube oil temperature is restored, the Narrow range level transmitter for the in-service channel fails low (off set). This is safety significant because it will require manual operation of the Feedwater System to select an operable Narrow Range level channel and return RPV level to normal.

When conditions have stabilized, RFPT A will trip and during the transient, Reactor Recirculation FCV B will fail to automatically runback. The RFPT A trip is safety significant because a reactor scram could result if the expected plant response is not verified. Failure of Reactor Recirculation FCV B to automatically runback is safety significant because Recirc Loop Flows will not be matched as required by Tech. Specs.

After level and power are stabilized, TBCC Pump A will trip and operator action will be required to manual start the standby TBCC Pump to allow for continued plant operation. Manual start of the standby TBCC Pump will also lead to a TBCC System piping leak that will grow progressively worse and require the crew to initiate a fast reactor shutdown.

Following the scram, a MSL pipe break in the Drywell occurs resulting in an MSIV isolation and rising Containment pressure. This is safety significant because PEI-B13, RPV Control (Non-ATWS,) and PEI-T23, Containment Control, must be entered and executed to maintain key Containment and RPV parameters.

When signaled to start on high Drywell pressure, RHR Pump A will not automatically start. This is safety significant because RHR Pump A must be manually started to assure it will perform its LPCI design function if required.

The RHR Pump A trip is safety significant because the crew must determine that RHR flow will be unavailable for containment spray, requiring RPV emergency depressurization to control the Containment pressure rise. The Reactor Level Transmitter downscale failure is safety significant because the failure could complicate RPV level control during and following the emergency depressurization.

Perry Plant
Initial License NRC examination
Operating Examination
Risk Significance

Each candidate will be examined in a dynamic simulator setting on a minimum of 2 events identified in the Perry IPE.

The dynamic simulator evaluation contains the following Perry IPE events.

Transient with a Loss of Power Conversion System:

- MSIV Isolation
- Various High Pressure Injection Systems unavailable
- RPV Emergency Depressurization required
- RPV level control with the ECCS low pressure systems

Intermediate LOCA Events:

- Steam Leak in Drywell
- Steam Break in Containment
- Failure of Long-Term Containment Heat Removal with RHR

ATWS:

- Failure to Insert All Control Rods
- Failure of RPV Level Control
- Failure of Standby Liquid Control (SLC) System
- RPV Emergency Depressurization required
- RPV level control with the ECCS low pressure systems

Additionally, two of four scenarios contain transient events that could lead to Perry IPE events if operator action is inappropriate or ineffective:

- IORV transient
- Trip of Service Water Pump

SRO candidates will be required to determine Risk related to equipment availability as part of the Administrative Examination.

PERRY

INITIAL LICENSE EXAM

MARCH 5 THRU 13, 2002

NRC Comments and Resolution on licensee submitted test outlines

Comments on the PERRY Exam Outlines

General	Comment(s)	
Systems Deleted:		
General	NRC:	Need a copy of the K/As/Systems deselected during the development of the outlines.
	LICENSEE RESPONSE:	*Per discussions with the licensee and the contractor hired to develop the exam outlines, no K/As were "deselected" prior to developing the outline.*
System 215001, "Traversing In-Core Probe" **REQUIRED CHANGE$_1$**	NRC:	Although licensed operators at Perry do not operate the system, knowledge of how the system interfaces with NIs, RPS, etc. would be testable knowledge. It may be appropriate to eliminate some K/As under this system but removing the entire system is not.
	LICENSEE RESPONSE:	*The licensee took action to reintroduce this system in the K/A selection process. A partial random re-selection was performed. This system was not selected. The system will be included in future outline development activities.*
System 290001, "Secondary Containment" **REQUIRED CHANGE$_2$**	NRC:	This system contains K/As that specifically refer to BWR-6 design. Although it may be appropriate to eliminate some K/As under this system, eliminating the entire system is not.
	LICENSEE RESPONSE:	*The licensee took action to reintroduce this system in the K/A selection process. A partial random re-selection was performed. This system was not selected. The system will be included in future outline development activities.*

Comments on the PERRY Exam Outlines

JPM/Scenario Event Number	Comment(s)	
A.2 (SRO) (perform risk assessment) Comment	NRC: LICENSEE RESPONSE:	The task should require the applicant to make an "SRO level" decision concerning the performance of maintenance. *The licensee understood and will ensure the JPM includes a verifiable action that allows the examiners to evaluate the applicant's ability to perform "SRO level" actions in regards to maintenance planning.*
All Scenarios	NRC: LICENSEE RESPONSE:	Although it appears that there are more than a sufficient number of events included in the scenarios, several of the events will be combined due to the difficulty in crediting individual actions once the event has been initiated. For example, in Scenario 1, Event 4, the RO would not be credited with an "N" for removing the RFPT from service, this is part and parcel to mitigating the event (RFPT bearing failure). *The licensee understood. This issue will be revisited during review of the "as submitted" exam material.*