Facility: Pilgrim Date of Exam: 2/26/2007

Tier	Group	RO K/A Category Points												SRO-Only Points		
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G*	Total	A2	G*	Total
1. Emergency & Abnormal Plant Evolutions	1	3	3	4	N/A			5	2	N/A		2	19	3	5	8
	2	1	2	2				1	1			1	8	2	2	4
	Tier Totals	4	5	6				6	3			3	27	5	7	12
2. Plant Systems	1	4	4	2	3	3	1	0	4	1	2	2	26	2	2	4
	2	1	0	0	0	1	2	2	0	2	2	2	12	0 0	2	2
	Tier Totals	5	4	2	3	4	3	2	4	3	4	4	38	2	4	6

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

Note:	1.	Ensure that at least two topics from every K/A category are sampled within each tier of the RO outline (i.e., the "Tier Totals" in each K/A category shall not be less than two). Refer to Section D.1.c for additional guidance regarding SRO sampling.
** See ES-401-4	2.	The point total for each group and tier in the proposed outline must match that specified in the table. The final point total for each group and tier may deviate by ± 1 from that specified in the table based on NRC revisions. The final RO exam must total 75 points and the SRO-only exam must total 25 points.
	3.	Select topics from many systems and evolutions; avoid selecting more than two K/A topics from a given system or evolution 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 (G) K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system. The SRO K/As must also be linked to 10 CFR 55.43 or an SRO-level learning objective.
	7.	On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings (IR) for the applicable license level, and the point totals for each system and category. Enter the group and tier totals for each category in the table above; summarize all the SRO-only knowledge and non-A2 ability categories in the columns labeled "K" and "A". Use duplicate pages for RO and SRO-only exams.
	8.	For Tier 3, enter the K/A numbers, descriptions, importance ratings, and point totals on Form ES-401-3.
	9.	Refer to ES-401, Attachment 2, for guidance regarding the elimination of inappropriate K/A statements.

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2007 NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

E/APE # / Name Safety Function	G	K1	K2	K3	A1	A2	Number	K/A Topic(s)	Imp.	Q#
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4	X						2.1.20	Conduct of Operations: Ability to execute procedure steps	4.2	76
295003 Partial or Complete Loss of AC / 6	X						2.1.12	Conduct of Operations: Ability to apply technical specifications for a system	4.0	77
295006 SCRAM / 1	X						2.1.23	Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.	4.0	78
295021 Loss of Shutdown Cooling / 4	X						2.4.4	Emergency Procedures / Plan Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.	4.3	79
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	80
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1						X	EA2.07	Ability to determine and/or interpret the following as they apply to SCRAM Condition Present and Power Above APRM Downscale or Unknown: Containment conditions/isolations	4.2	81
295028 High Drywell Temperature / 5	X						2.1.20	Conduct of Operations: Ability to execute procedure steps.	4.2	82
295030 Low Suppression Pool Water Level / 5						X	EA2.03	Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Reactor pressure	3.9	83
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4						X	AA2.05	Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Jet pump operability: Not-BWR-1&2	3.1	39
295003 Partial or Complete Loss of AC / 6					X		AA1.03	Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Systems necessary to assure safe plant shutdown	4.4	40
295004 Partial or Total Loss of DC Pwr / 6			X				AK2.03	Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: D.C. bus loads	3.3	41
295005 Main Turbine Generator Trip / 3		X					AK1.03	Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP : Pressure effects on reactor level	3.5	42

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2007 NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

E/APE # / Name Safety Function	G	K1	K2	K3	A1	A2	Number	K/A Topic(s)	Imp.	Q#
295006 SCRAM / 1			X				AK2.05	Knowledge of the interrelations between SCRAM and the following: CRD mechanism	3.1	43
295016 Control Room Abandonment / 7				X			AK3.03	Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT : Disabling control room controls	3.5	44
295018 Partial or Total Loss of CCW / 8				X			AK3.05	Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Placing standby heat exchanger in service	3.2	45
295019 Partial or Total Loss of Inst. Air / 8				X			AK3.02	Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Standby air compressor operation	3.5	46
295021 Loss of Shutdown Cooling / 4		X					AK1.03	Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : Adequate core cooling	3.9	48
295032 HI Secondary Containment Area Temps / 8			X				EK2.01	Knowledge of the interrelations between HI SECONDARY CONTAINMENT AREA TEMP and the following: Area/room coolers	3.4	49
295024 High Drywell Pressure / 5					X		EA1.12	Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE: Suppression pool spray: Mark-I&II	3.8	50
295025 High Reactor Pressure / 3		X					EK1.01	Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : Pressure effects on reactor power	3.9	51
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	52
295028 High Drywell Temperature / 5						X	EA2.03	Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Reactor water level	3.7	53
295030 Low Suppression Pool Water Level / 5				X			EK3.01	Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: Emergency depressurization	3.8	54
295031 Reactor Low Water Level / 2	X						2.2.22	Equipment Control Knowledge of limiting conditions for operations and safety limits.	3.4	55

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2007 NRC Written Examination Outline
Emergency and Abnormal Plant Evolutions – Tier 1 Group 1

E/APE # / Name Safety Function	G	K1	K2	K3	A1	A2	Number	K/A Topic(s)	Imp.	Q#
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1	X						2.4.6	Emergency Procedures / Plan Knowledge symptom based EOP mitigation strategies.	3.1	56
295038 High Off-site Release Rate / 9					X		EA1.01	Ability to operate and/or monitor the following as they apply to high off-site release rate: Stack gas monitoring system.	3.9	57
600000 Plant Fire On-site / 8					X		AA1.06	Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Fire alarm	3.0	58
K/A Category Point Totals:	2/5	3	3	4	5	2/3	Group Point Total:			**19/8**

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Emergency and Abnormal Plant Evolutions – Tier 1 Group 2

E/APE # / Name Safety Function	G	K1	K2	K3	A1	A2	Number	K/A Topic(s)	Imp.	Q#
295010 High Drywell Pressure / 5	X						2.4.30	Emergency Procedures / Plan Knowledge of which events related to system operations/status should be reported to outside agencies.	3.6	84
295012 High Drywell Temperature / 5						X	AA2.01	Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Drywell temperature	3.9	85
295029 High Suppression Pool Water Level / 5	X						2.2.25	Equipment Control Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.	3.7	86
500000 High CTMT Hydrogen Conc. / 5						X	EA2.03	Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Combustible limits for drywell	3.8	87
295007 High Reactor Pressure / 3	X						2.4.49	Emergency Procedures/Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls	4.0	47
295009 Low Reactor Water Level / 2			X				AK2.03	Knowledge of the interrelations between LOW REACTOR WATER LEVEL and the following: Recirculation system	3.1	59
295013 High Suppression Pool Temperature / 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	60
295014 Inadvertent Reactivity Addition / 1			X				AK2.01	Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following: RPS	3.9	61
295015 Incomplete SCRAM / 1						X	AA2.02	Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM : Control rod position	4.1	62
295020 Inadvertent Cont. Isolation / 5 & 7				X			AK3.03	Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION: Drywell/containment temperature response	3.2	63
295022 Loss of CRD Pumps / 1				X			AK3.01	Knowledge of the reasons for the following responses as they apply to LOSS OF CRD PUMPS: Reactor SCRAM	3.7	64
295029 High Suppression Pool Water Level / 5		X					EK1.01	Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Containment integrity	3.4	65

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Emergency and Abnormal Plant Evolutions – Tier 1 Group 2

E/APE # / Name Safety Function	G	K1	K2	K3	A1	A2	Number	K/A Topic(s)	Imp.	Q#
K/A Category Point Total:	1/2	1	2	2	1	1/2	Group Point Total:			8/4

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2007 NRC Written Examination Outline
Plant Systems – Tier 2 Group 1

System #/Name	G	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	Number	K/A Topics	Imp.	Q#
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	88
215004 Source Range Monitor	X											2.2.25	Equipment Control Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.	3.7	89
261000 SGTS									X			A2.03	Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High train temperature	3.2	90
262001 AC Electrical Distribution	X											2.1.12	Conduct of Operations: Ability to apply technical specifications for a system.	4.0	91
203000 RHR/LPCI: Injection Mode			X									K2.03	Knowledge of electrical power supplies to the following: Initiation logic	2.7	1
205000 Shutdown Cooling				X								K3.03	Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor temperatures (moderator, vessel, flange)	3.8	2
206000 HPCI											X	A4.01	Ability to manually operate and/or monitor in the control room: Turbine speed controls	3.8	3
206000 HPCI			X									K2.01	Knowledge of electrical power supplies to the following: System valves: BWR-2,3,4	3.2	4
209001 LPCS		X										K1.08	Knowledge of the physical connections and/or cause- effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: A.C. electrical power	3.2	5
211000 SLC		X										K1.02	Knowledge of the physical connections and/or cause- effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following: Core plate differential pressure indication	2.7	6
211000 SLC											X	A4.01	Ability to manually operate and/or monitor in the control room: Tank level	3.9	7

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2007 NRC Written Examination Outline
Plant Systems – Tier 2 Group 1

System #/Name	G	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	Number	K/A Topics	Imp.	Q#
212000 RPS					X							K4.08	Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Complete control rod insertion following SCRAM signal generation	4.2	8
212000 RPS						X						K5.02	Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM :Specific logic arrangements	3.3	9
215003 IRM					X							K4.04	Knowledge of INTERMEDIATE RANGE MONITOR (IRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Varying system sensitivity levels using range switches	2.9	10
215004 Source Range Monitor			X									K2.01	Knowledge of electrical power supplies to the following: SRM channels/detectors	2.6	11
215005 APRM / LPRM										X		A3.03	Ability to monitor automatic operations of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM including: Meters and recorders	3.3	12
217000 RCIC									X			A2.09	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: Loss of Vacuum Pump	2.9	13
218000 ADS	X											2.1.27	Conduct of Operations: Knowledge of system purpose and or function.	2.8	14
223002 PCIS/Nuclear Steam Supply Shutoff							X					K6.06	Knowledge of the effect that a loss or malfunction of the following will have on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF : Various process instrumentation	2.8	15
239002 SRVs	X											2.2.25	Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operation and safety limits	2.5	16
239002 SRVs						X						K5.01	Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES : Relief function of SRV operation	3.4	17

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2007 NRC Written Examination Outline
Plant Systems – Tier 2 Group 1

System #/Name	G	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	Number	K/A Topics	Imp.	Q#
259002 Reactor Water Level Control					X							K4.12	Knowledge of REACTOR WATER LEVEL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Manual and automatic control of the system	3.5	18
261000 SGTS									X			A2.12	Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High Fuel Pool ventilation radiation: Plant-Specific	3.2	19
262001 AC Electrical Distribution		X										K1.03	Knowledge of the physical connections and/or cause- effect relationships between A.C. ELECTRICAL DISTRIBUTION and the following: Off-site power sources	3.4	20
262002 UPS (AC/DC)									X			A2.01	Ability to (a) predict the impacts of the following on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Under voltage	2.6	21
263000 DC Electrical Distribution			X									K2.01	Knowledge of electrical power supplies to the following: Major D.C. loads	3.1	22
264000 EDGs				X								K3.03	Knowledge of the effect that a loss or malfunction of the EMERGENCY GENERATORS (DIESEL/JET) will have on following: Major loads powered from electrical buses fed by the emergency generator(s)	4.1	23
300000 Instrument Air						X						K5.01	Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Air compressors	2.5	24
400000 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 operation: High/low CCW temperature	2.9	25
400000 Component Cooling Water		X										K1.02	Knowledge of the physical connections and / or cause-effect relationships between CCWS and the following: Loads cooled by CCWS	3.2	26
K/A Category Point Totals:	2/2	4	4	2	3	3	1	0	4/2	1	2	Group Point Total:			**26/4**

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2007 NRC Written Examination Outline
Plant Systems – Tier 2 Group 2

System #/Name	G	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	Number	K/A Topics	Imp.	Q#
204000 RWCU	X											2.4.28	Knowledge of procedures relating to emergency response to sabotage.	3.3	92
286000 Fire Protection	X											2.1.23	Ability to perform specific and integrated plant procedures during all modes of plant operation	4.0	93
201002 RMCS	X											2.1.28	Conduct of Operations: Knowledge of purpose and function of major system components and controls	3.2	27
239001 Main and Reheat Steam						X						K5.05	Knowledge of the operational implications of the following concepts as they apply to Main and Reheat Steam system: Flow Indication	2.8	28
201006 RWM								X				A1.01	Ability to predict and/or monitor changes in parameters associated with operating the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) controls including: Rod position: P-Spec(Not-BWR6)	3.2	29
204000 RWCU							X					K6.09	Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER CLEANUP SYSTEM : CRD hydraulics: Plant-Specific	2.7	30
215001 Traversing In-core Probe											X	A4.03	Ability to manually operate and/or monitor in the control room: Isolation valves: Mark-I&II(Not-BWR1)	3.0	31
215002 RBM										X		A3.01	Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including: Four rod display: BWR-3,4,5	3.1	32
216000 Nuclear Boiler Inst.	X											2.2.25	Equipment Control Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.	2.5	33
226001 RHR/LPCI: CTMT Spray Mode		X										K1.11	Knowledge of the physical connections and/or cause- effect relationships between RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE and the following: Component cooling water systems	2.8	34
241000 Reactor/Turbine Pressure Regulator										X		A3.09	Ability to monitor automatic operations of the REACTOR/TURBINE PRESSURE REGULATING SYSTEM including: Control/governor valve operation	3.3	35
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	36

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Plant Systems – Tier 2 Group 2

System #/Name	G	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	Number	K/A Topics	Imp.	Q#
272000 Radiation Monitoring							X					K6.03	Knowledge of the effect that a loss or malfunction of the following will have on the RADIATION MONITORING SYSTEM : AC power	2.8	37
228000 Plant Ventilation											X	A4.01	Ability to operate and/or monitor in the control room: Start and Stop fans	3.1	38
K/A Category Point Totals:	2/2	1	0	0	0	1	2	2	0/0	2	2	Group Point Total:			**12/2**

Facility:	Pilgrim		Date of Exam:	2/26/2007			
Category	K/A #	Topic	RO		SRO-Only		
			IR	Q#	IR	Q#	
1. Conduct of Operations	2.1.11	Knowledge of less than one hour technical specification action statements for systems.			3.8	94	
	2.1.18	Ability to make accurate, clear and concise logs, records, status boards, and reports.	2.9	66			
	2.1.8	Ability to coordinate personnel activities outside the control room.	3.8	67			
	2.1.25	Ability to obtain and interpret station reference materials such as graphs, monographs, and tables which contain performance data.	2.8	68			
	2.1.2	Knowledge of operator responsibilities during all modes of plant operation.	3.0	69			
	Subtotal			4		1	
2. Equipment Control	2.2.25	Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.			3.7	95	
	2.2.32	Knowledge of the effects of alterations on core configuration.			3.3	96	
	2.2.27	Knowledge of the refueling process.	2.6	70			
	2.2.34	Knowledge of the process for determining the internal and external effects on core reactivity.	2.8	71			
	Subtotal			2		2	
3. Radiation Control	2.3.4	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.			3.1	97	
	2.3.9	Knowledge of the process for performing a containment purge.			3.4	98	
	2.3.11	Ability to control radiation releases.	2.7	72			
	2.3.1	Knowledge of 10CFR:20 and related facility radiation control requirements	2.6	73			
	Subtotal			2		2	
4. Emergency Procedures / Plan	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	99	
	2.4.10	Knowledge of annunciator response procedures.			3.1	100	
	2.4.5	Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.	2.9	74			
	2.4.2	Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	3.9	75			
	Subtotal			2		2	
Tier 3 Point Total				10		7	

Tier / Group	Randomly Selected K/A	Reason for Rejection
2 / 2	201005 K5.02	System does not exist at facility. Randomly reselected 239001 K5.05
1 / 1	295028 G2.1.27	Impossible to meet KA Topic requirement at SRO level. Randomly reselected G2.1.20 for APE.
2 / 1	262001 G2.1.30	Impossible to meet KA Topic requirement at SRO level. Randomly reselected G2.1.12 for system.
2 / 2	272000 K6.01	No effect between systems, either directly or indirectly. Randomly reselected K6.03 for system
2 / 1	206000 A4.11	Component does not exist at facility. Randomly reselected A4.03 for system. Subsequently randomly selected A4.01 for same reason.
2 / 1	212000 K5.01	Concept does not apply to facility. Randomly reselected K5.02 for system
2 / 1	217000 A2.17	Action does not exist at facility. Randomly reselected A2.09 for topic
2 / 1	261000 A2.14	No action for condition at facility. Randomly reselected A2.12 for topic
1 / 1	295005 AK1.02	Not applicable to facility. Randomly reselected AK1.03 for topic
1 / 1	295024 EA2.03	Impossible to meet topic requirement at SRO Level. Randomly reselected EA2.01 for topic
2 / 2	286000 G2.2.26	Impossible to develop question applicable to facility. Randomly reselected G2.1.23 for topic
1 / 1	**295019 2.1.14	Topic area not directly or indirectly pertinent to system. Randomly reselected 295007 2.4.49 4.0 (Q. #47) ***This moved a T1 G1 topic to a T1 G2 topic***
1 / 1	295001 G2.1.28	Impossible to meet KA Topic requirement at SRO level. Randomly reselected G2.1.20 (4.2) for APE. (Q. #76)
1 / 1	295005 AA.2.04	Impossible to meet KA requirement at SRO level. Randomly reselected 295003 G2.1.12 (4.0) for system. (Q. #77)
2 / 2	201002 G2.1.14	Topic area not directly or indirectly pertinent to system. Randomly reselected G2.1.28 (3.2) for system. (Q. #27)
1 / 1	295025 EA.2.04	Double jeopardy with question #83. Randomly reselected 295037 EA2.07 (4.2) for system. (Q. #81)
1 / 1	**295038 EK1.01	No operational valid question could be written for the topic. Randomly reselected EA 1.01 for topic (Q. #57) ***This moved a T1 K1 to a T1 A1 topic***
1 / 1	295023 AK2.03	Double jeopardy with Q. #19. Randomly reselected 295032 EK2.01 (3.5) for Q.#49
1 / 1	**204000 A2.11	Operational valid SRO level question could not be written for topic. Randomly reselected G2.4.28 (3.3) (Q. #92) *This moved a T2 A2 to a T2 G topic*

Tier / Group	Randomly Selected K/A	Reason for Rejection
2 / 2	201005 K5.02	System does not exist at facility. Randomly reselected 239001 K5.05
1 / 1	295028 G2.1.27	Impossible to meet KA Topic requirement at SRO level. Randomly reselected G2.1.20 for APE.
2 / 1	262001 G2.1.30	Impossible to meet KA Topic requirement at SRO level. Randomly reselected G2.1.12 for system.
2 / 2	272000 K6.01	No effect between systems, either directly or indirectly. Randomly reselected K6.03 for system
2 / 1	206000 A4.11	Component does not exist at facility. Randomly reselected A4.03 for system
2 / 1	212000 K5.01	Concept does not apply to facility. Randomly reselected K5.02 for system
2 / 1	217000 A2.17	Action does not exist at facility. Randomly reselected A2.09 for topic
2 / 1	261000 A2.14	No action for condition at facility. Randomly reselected A2.12 for topic
1 / 1	295005 AK1.02	Not applicable to facility. Randomly reselected AK1.03 for topic

Facility: Pilgrim	Date of Examination: 2/26/2007
Examination Level (circle one): RO	Operating Test Number: NRC

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M	JPM – Perform a Short Form Heat Balance K/A: 2.1.7 (3.7) Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation.
Conduct of Operations	N	JPM – Verify AOG Recombine Operation K/A: 2.2.13 (3.8) Ability to obtain and interpret station reference materials such as graphs / monographs / and tables which contain performance data.
Equipment Control	N	JPM: Prepare an SLC Pump tagging clearance K/A: 2.2.13 (3.0) Knowledge of tagging and clearance procedures (3.6).
Radiation Control	N	JPM: Determine Stay Time. K/A: 2.3.2 (2.5) Knowledge of facility ALARA program
Emergency Plan		

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

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

A.1.a The candidate will perform a short form heat balance by collecting the required plant data from control room indications and performing a manual calculation. The critical task will be to accurately calculate reactor thermal power. This is a modified bank JPM.

A.1.b The candidate will be required to identify improper recombiner operation by collecting the required plant data from control room indications and utilizing the appropriate attachments of PNPS 2.4.141, Abnormal Recombiner Operation. Plant data will indicate recombiner outlet temperature is not excessively high; however, abnormal recombiner delta-T will be indicated. The critical task will be to determine the recombiner is overheated at a reduced power level. This is a new JPM.

A.2 The candidate will prepare an SLC Pump tagging clearance. The critical task will be to prepare a tagout that assures equipment and personal safety. This is a new JPM.

A.3 The candidate will determine stay time given a set of plant conditions. The candidate must have knowledge of the facility limits to perform the calculation. The critical task will be to determine the correct stay time. This is a new JPM.

Facility: Pilgrim		Date of Examination:	2/26/07
Examination Level (circle one): **SRO**		Operating Test Number:	NRC

Administrative Topic (see Note)	Type Code*	Describe activity to be performed
Conduct of Operations	M	JPM – Perform a Short Form Heat Balance K/A: 2.1.7 (4.4) Ability to evaluate plant performance and make operational judgments based on operating characteristics / reactor behavior / and instrument interpretation.
Conduct of Operations	N	JPM – Verify AOG Recombiner Operation K/A: 2.1.25 (3.1) Ability to obtain and interpret station reference materials such as graphs / monographs / and tables which contain performance data.
Equipment Control	N	JPM: Review an SLC Pump tagging clearance K/A: 2.2.13 (3.8) Knowledge of tagging and clearance procedures
Radiation Control	N	JPM: Determine Stay Time. K/A: 2.3.2 (2.9) Knowledge of facility ALARA program
Emergency Plan	M	JPM: Perform Dose Assessment Using DAPAR Software K/A: 2.4.44 (4.0) Knowledge of emergency plan protective action recommendations

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

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

A.1.a The candidate will perform a short form heat balance by collecting the required plant data from control room indications and performing a manual calculation. The critical task will be to calculate the correct reactor thermal power. This is a modified bank JPM.

A.1.b The candidate will be required to identify improper recombiner operation by collecting the required plant data from control room indications and utilizing the appropriate attachments of PNPS 2.4.141, Abnormal Recombiner Operation. Plant data will indicate recombiner outlet temperature is not excessively high; however, abnormal recombiner delta-T will be indicated. The critical task will be to determine the recombiner is overheated at a reduced power level. This is a new JPM.

A.2 The candidate will review an SLC Pump tagging clearance. The critical task will be to identify all tagging errors. This is a new JPM.

A.3 The candidate will determine stay time given a set of plant conditions. The candidate must have knowledge of the facility limits to perform the calculation. The critical task will be to determine the correct stay time. This is a new JPM.

A.4 The candidate will perform a dose assessment using the DAPAR Software. The critical task will be to accurately perform the dose assessment and determine the appropriate protective action recommendations. This is a modified JPM.

Facility: PNPS		Date of Examination:	2/26/2007
Exam Level (circle one): **RO** / **SRO(I)** / SRO (U)		Operating Test No.:	

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

	System / JPM Title	Type Code*	Safety Function
S-1	201003 Control Rod and Drive Mechanism (JPM-201-11) Reactor Startup to Criticality	D, A, L, S	1
S-2	217000 Reactor Core Isolation Cooling System (JPM-217-03a) Inject to RPV with RCIC	M, A, S	2
S-3	262001 A.C. Electrical Distribution (JPM-262-10) Restoration of Power to 4160 VAC Bus A5 from SUT	D, S	6
S-4	206000 High Pressure Coolant Injection System (JPM-206-09) Operate HPCI for pressure Control	M, A, S	4
S-5	400000 Component Cooling Water System (JPM-200-31) Recover RBCCW Loop B with an elevated Drywell Temperature	M, A, S	8
S-6	239001 Main and Reheat Steam System (JPM-200-05) Respond to an MSIV Closure	D, S	3
S-7	261000 Standby Gas Treatment System (JPM-229-012) Manually Start SBGT and Vent the Torus	D, A, S	9
S-8	212000 Reactor Protection System (JPM-212-04) Reset a Reactor Scram **(RO only)**	D, S	7

In-Plant Systems@ (3 for RO; 3 for SRO-I; 3 or 2 for SRO-U)

	System / JPM Title	Type Code*	Safety Function
P-1	212000 Reactor Protection System (JPM-200-22) Reactor Scram from Outside Control Room	D, E	7
P-2	239001 Main and Reheat Steam System (JPM-200-16) Defeat MSIV Isolation Signals	D, E	3
P-3	201001 Control Rod Drive Hydraulic System (JPM-201-03) Shift CRD Flow Control Valves	D, R	1

@	All RO and SRO-I control room (and in-plant) systems must be different and serve different safety functions; all 5 SRO-U systems must serve different safety functions; in-plant systems and functions may overlap those tested in the control room.
* Type Codes	Criteria for RO / SRO-I / SRO-U
(A)lternate path	4-6 / 4-6 / 2-3
(C)ontrol room	
(D)irect from bank	≤ 9 / ≤ 8 / ≤ 4
(E)mergency or abnormal in-plant	≥ 1 / ≥ 1 / ≥ 1
(L)ow-Power / Shutdown	≥ 1 / ≥ 1 / ≥ 1
(N)ew or (M)odified from bank including 1(A)	≥ 2 / ≥ 2 / ≥ 1
(P)revious 2 exams	≤ 3 / ≤ 3 / ≤ 2 (randomly selected)
(R)CA	≥ 1 / ≥ 1 / ≥ 1
(S)imulator	

NRC JPM Examination
Summary Description

S-1 The candidate will continue a reactor startup, withdrawing control rods on a slightly subcritical reactor including the performance of CRDM coupling checks. The alternate path requires that the candidate recognize indications of an uncoupled control rod and take actions in accordance with PNPS 2.4.11, Control Rod Positioning Malfunctions. This is a bank JPM in the CRD system - Reactivity Control Safety Function.

S-2 The candidate will place RCIC in injection mode and raise reactor water level. The alternate path requires that the candidate recognize a failure of the cooling water supply valve to reposition automatically and take manual action to open the valve. This is a modified JPM in the RCIC system - Reactor Water Inventory Control Safety Function.

S-3 The candidate will perform a dead bus transfer and transfer the A5 bus back to the startup transformer during a loss of Off-Site power. This is a bank JPM under the AC Electrical Distribution – Electrical Systems Safety Function.

S-4 The candidate will start HPCI in pressure control mode. The alternate path requires that the candidate recognize a failure of the HPCI flow controller low and place the controller in manual and raise flow in to establish HPCI operation in pressure control. This is a modified JPM in the HPCI system - Heat Removal From Reactor Core Safety Function.

S-5 The candidate will take action to restore RBCCW Loop 'B' system flow following a non-LOCA event that resulted in a reactor scram and elevated drywell temperatures. The alternate path requires that the candidate recognize a breach in the RBCCW System piping inside the drywell and isolate RBCCW system flow. This is a modified JPM in the Reactor Building Component Cooling Water System - Plant Service Systems Safety Function.

S-6 The candidate is required to recognize an MSIV closure, enter the appropriate procedures, and open the steam line drains. This is a bank JPM under the Main and Reheat Steam System – Reactor Pressure Control Safety Function.

S-7 The candidate is required to place the "A" train of SBGT in service to vent the primary containment through the Torus. The alternate path requires that the candidate terminate venting when an alarm is received which requires termination of venting. This is a bank JPM in the Standby Gas Treatment System - Radioactivity Release Safety Function.

S-8 The candidate is required to bypass SDIV scram, reset scram, wait for SDIV to drain, and return the bypass switch to normal. This is a bank JPM in the Reactor Protection System - Instrumentation Safety Function.

P-1 The candidate is required to initiate a reactor scram from outside of the control room. This is a bank JPM in the Reactor Protection System - Instrumentation Safety Function.

P-2 The candidate is required to defeat main steam isolation signals to facilitate venting the reactor pressure vessel. This is a bank JPM in the Main and Reheat Steam System - Reactor Pressure Control Safety Function.

P-3 The candidate is required to shift CRD Flow Control Valves. This is a bank JPM in the Control Rod Drive Hydraulic system - Reactivity Control Safety Function.

Facility: Pilgrim	Scenario No.: 1	Op Test No.: 2007	
Examiners:		Operators:	

Initial Conditions:

Power was lowered to 90% for control rod pattern adjustment which has been completed

HPCI OOS for Aux Oil Pump Replacement – 14 Day LCO

RBCCW Pump A (P-202A) OOC - Tracking LCO

IRM G is bypassed.

LPRM 36-13-B is bypassed

Turnover:

A CRD Suction filter D/P is high. Replace filter IAW 2.2.87, Attachment 2.

Return power to 100% using recirc flow

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – RO, SRO	Swap CRD Pumps for suction filter replacement
2	N/A	R - RO	Raise reactor power with Recirc
3	FW09	I – RO, SRO	Master FWLC fails as is
4	CW05B	I – BOP, SRO TS-SRO	RBCCW "B" pump trip with failure of standby pump to auto start
5	FW23	I-RO, SRO TS-SRO	FWLC NR Channel B fails high
6	FW21B	C – RO, SRO	Condensate pump B trips.
7	ED06	M - All	Loss of all offsite power
	I/O	C – BOP	RCIC Injection Mode Push Button Fails RCIC auto initiation fails
8	PC01	M - All	Recirc leak in Drywell leads to Emergency Depressurization on low RPV level
9	CS02A	C- BOP	"A" CS Injection valve (25A) failure to auto open
10	CS02B	C- BOP	"B" CS Injection valve (25A) failure to auto open
11	RH04B	C- BOP	"B" RHR LPCI injection valve failure to auto open (29B)

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

Scenario 1 Description

Power was reduced to 90% for a control rod pattern adjustment. Due to CRD Pump suction filter high D/P, the FSS will request the crew to swap CRD Pumps. After CRD pumps have been swapped, the crew will raise reactor power with recirc as directed in the shift turnover.

When raising power with recirc, the FWLC Master Controller will fail as is requiring the crew to enter procedure 2.4.49, Feedwater Malfunctions and place the Master FWLC Controller in manual.

After level is stable with the FWLC Master Controller in manual, RBCCW pump "B" will trip and the standby pump will fail to auto start requiring operator action to start the standby RBCCW pump.

After Tech Specs are evaluated for RBCCW, FWLC Narrow Range Channel B fails high, requiring the crew to swap to Narrow Range Channel A and evaluate Tech Specs.

When Tech Specs have been referenced for the narrow range transmitter failure, Condensate Pump B will trip resulting in a trip of the B RFP and an automatic Recirc runback when RPV drops to 19 inches. Because the FWLC Master Controller is in manual, the crew must closely monitor and control RPV level during the Recirc runback.

When conditions have stabilized following the Condensate Pump trip and recirc runback, a loss of all offsite power will occur resulting in a loss of all feedwater and a reactor scram. The crew will enter and execute EOP-1. Following the scram, RCIC will fail to initiate automatically, and the RCIC Injection Mode Push Button will also fail, requiring the crew to manually align RCIC and inject to the vessel. The crew will also maximize injection with the available CRD pump.

Once conditions have stabilized post scram, a recirc leak will develop in the drywell and drywell pressure will rise, requiring EOP-3 entry and EOP-1 re-entry. Torus cooling, torus spray, and drywell spray will be directed in accordance with EOP-3. As required by EOP-1, the crew will maximize flow with available high pressure injection systems; however, the recirc leak in Drywell will continue to worsen and vessel level will lower until Emergency Depressurization and injection with low pressure ECCS systems is required. When all SRVs have been opened and the low RPV pressure valve permissives are received, both Core Spray injection valves and the B LPCI injection valve will fail to open automatically requiring manual operator action to open the valves and recover vessel level.

Facility: Pilgrim		Scenario No.: 2	Op Test No.: 2007
Examiners:		Operators:	

Initial Conditions: Power is 2.5% with a startup in progress

2.2.96, Attachment 15, preset checks are done on all feedpumps

Two Condensate pumps in service. Running additional condensate pump to wear new bearing in

IRM "G" is bypassed

Turnover:

Continue startup. At 5% power, transfer RMS to run and resume pulling control rods.

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	R – RO N - SRO	Pull rods to continue power ascension Transfer RMS to run
2	NM21E	I – RO, SRO TS-SRO	APRM 'E' fails downscale
3	RD02	C – RO, SRO	In service CRD flow control valve fails closed
4	RD0 22-23	C – RO, SRO TS - SRO	Control rod 22-23 drifts inward
5	N/A	N - BOP	RFP 'C' intermittent TBCCW leak, place RFP 'B' in service
6		I – BOP, SRO	RWCU Pump 'A' RBCCW Temp High, pump fails to auto trip
7	PC02 RM07	M - All	RWCU leak leading to scram
8	RP14A	I - RO	Manual scram failure. ARI required.
9	RM07	M - All	RWCU leak leads to Emergency Depressurization
10	RH04B	C - BOP	SRV 'B' fails to open

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

Scenario 2 Description

The crew will take the watch with a reactor startup in progress. They will withdraw control rods, continuing the startup until 5% power is obtained and the reactor mode switch is placed in run.

After the mode switch is in run, APRM E will fail downscale. The APRM will be bypassed, and after Tech Specs have been referenced, the startup will continue.

When the startup continues, the crew will discover that the in-service CRD Flow Control Valve has failed closed requiring action to remove the failed FCV from service and place the standby CRD Flow Control Valve in service.

After placing the standby CRD Flow Control Valve in service, the startup will continue until control rod 22-23 drifts inward. The crew will take actions in accordance with PNPS 2.4.11, Control Rod Positioning Malfunctions and address Tech Specs for the inoperable control rod. The startup will be halted.

While waiting for troubleshooting and reactor maneuvering plans, an intermittent TBCCW leak will develop on RFP C requiring the crew to place RFP B in service and secure RFP C in accordance with PNPS 2.2.96, Condensate and Feedwater.

After RFP B in service, an RBCCW high temperature condition will develop on the "A" RWCU Pump. RWCU pump "A" will fail to automatically trip and the crew will take action in accordance with the ARP to manually stop the pump.

When RWCU Pump "A" is stopped, a leak will develop on the RWCU pump and temperatures will rise in the RWCU pump room requiring entry into EOP-4. RWCU area temperatures will continue to rise until a manual scram is required, and the crew will enter and execute EOP-1. When manual scram is attempted, the scram push buttons and reactor mode switch will fail to initiate rod movement; however, all control rods will insert when the ARI pushbuttons are depressed.

Following control rod insertion, RWCU area temperatures will continue to rise until conditions are degraded in two of the areas specified in EOP-4 and Emergency Depressurization is required. The crew will execute EOP-17; however, one SRV will fail to open requiring the crew to utilize Alternate RPV Depressurization Systems (SRV Remote SD Panel) to augment emergency depressurization.

Facility: Pilgrim		Scenario No.: 3	Op Test No.: 2007
Examiners:		Operators:	

Initial Conditions: Power is 50%

HPCI OOS for Aux Oil Pump Replacement – 14 Day LCO

RBCCW Pump A (P-202A) OOC, Tracking LCO

IRM G is bypassed.

LPRM 36-13-B is bypassed.

Turnover:

Remove Seawater pump "B" from service and perform emergency backwash

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N – BOP, SRO	Remove Seawater pump "B" from service for emergency backwash
2	RR11A	C – RO, SRO	'A' RRP Pump Motor Vibration High
3	NM12A	I – RO, SRO TS - SRO	'A' IRM fails downscale
4	RD05A	C- RO, SRO	'B' CRD Pump Trip
5	RR13A RR13B	C – RO, SRO	'A' RRP Inner Seal Failure. A RRP Outer Seal Failure
6	N/A	R – RO, SRO	Insert control rods to exit the exclusion region
7	TC06	C – BOP, SRO TS-SRO	RCIC Steam Leak, failure to auto isolate.
8	TC01	M - All	Main Turbine Trip, ATWS.
9		C - RO	First squib valve fails to open when fired
10		C - RO	RWCU MO80 fails to isolate automatically

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

Scenario 3 Description

After taking the watch, as directed in the shift turnover, the crew will remove Seawater pump "B" from service for emergency backwash of the inlet water boxes.

After backwash is in progress, the crew will respond to a Recirc Pump Motor Vibration High alarm. Recirc Pump speed will be lowered as directed by the ARP. When Recirc pump speed has been reduced a small amount, the Recirc Pump Motor Vibration Monitor will reset.

After the Recirc Pump Motor Vibration High alarm is reset, the "A" IRM will fail downscale. The crew will take actions in accordance with the ARP, and with IRM 'G' previously inoperable and bypassed, a Tech Spec LCO will be entered.

After Tech Specs have been evaluated, the B CRD Pump will trip. The crew will take actions per PNPS 2.4.4, Loss of CRD Pumps, including immediate action to verify reactor pressure is greater than 950 psig. When directed, the RO will place the standby CRD Pump in service.

When the standby CRD Pump is in service, the 'A' Recirc Pump inner and outer seals will fail in sequence. Initially, the inner seal will fail requiring action to monitor seal status and drywell conditions per the ARP as well as entry into PNPS 2.4.22. After a brief period of time, the outer seal will fail. When the crew determines a catastrophic seal failure has occurred, the 'A' Recirc Pump will be tripped and isolated.

After the 'A' Recirc Pump is tripped, the crew will estimate total core flow, plot the location on Power/Flow Map, and take actions in accordance with PNPS 2.4.17, Recirc Pump Trip and PNPS 2.4.165, Thermal Hydraulic Instabilities. Control rods will be manually inserted to exit the exclusion region. The Reactor Operator is expected to closely monitor LPRM indications for indications of thermal hydraulic instability.

When control rods have been inserted sufficiently, a steam leak will develop in the RCIC steam line, and RCIC will fail to isolate automatically. The crew is expected to take action to isolate RCIC manually; additionally, the CRS will enter and direct actions per EOP-4 which will direct RCIC isolation if action has not been taken previously. When RCIC is isolated (and with HPCI already inoperable), the CRS will evaluate Tech Specs and enter a 24 hour LCO.

After Tech Specs have been addressed, the main turbine will trip and an ATWS will result due to hydraulic lock of the scram discharge volume. The CRS is expected to enter and direct actions per EOP-02, including SLC initiation. When the SLC Actuate switch is placed in the SYS 'A' or SYS 'B' position, the selected SLC pump will start; however, the associated squib valve will fail to open as indicated by high SLC pump discharge pressure and zero flow. The reactor operator is expected to select the alternate train and inject SLC. Additionally, the RWCU return isolation valve (MO80) will fail to auto isolate, requiring manual operator action to isolate the RWCU system. ATWS actions will continue per EOP-02 until all rods have been fully inserted; EOP-01 has been entered, and reactor level is in the normal band.

Facility: Pilgrim		Scenario No.: 4	Op Test No.: 2007
Examiners:		Operators:	

Initial Conditions: Power is 100%

HPCI OOS for Aux Oil Pump Replacement – 14 Day LCO

RBCCW Pump A (P-202A) OOC, Tracking LCO

LPRM 36-13-B is bypassed

IRM G is bypassed.

Turnover:

Shift TBCCW Pumps for maintenance vibration Test

Lower power for control rod pattern adjustment

Event No.	Malf. No.	Event Type*	Event Description
1	N/A	N - BOP	Shift TBCCW Pumps for maintenance vibration Test
2	N/A	R - RO	Reduce power with Recirc
3	NM17 36-45-B	I – RO, SRO TS – SRO	LPRM 36-45-B fails upscale
4	TC06	I BOP, SRO TS – SRO	EPR Pressure Oscillations
5	RR20A	I – RO, SRO TS - SRO	“B” Recirc Flow Controller fails upscale
6	MS14D	M - All	D SRV fails open, Manual reactor scram
7	RH04B	C- BOP	PASS H2/O2 Sample valve fails to Isolate (CV91)
8		C - BOP	RBCCW to A RHR HX inlet valve fails shut
9	PC22	M - All	D SRV tail pipe fails leading to Emergency Depressurization

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

Scenario 4 Description

After taking the watch, as directed in the shift turnover, the crew will shift TBCCW pumps. After the TBCCW pumps have been swapped, the crew will proceed with a planned power reduction using Recirc flow.

While the planned power reduction is underway, LPRM 36-45-B will fail upscale, and the CRS will enter and direct actions per PNPS 2.4.38, LPRM Failure. The crew will bypass the failed LPRM and verify that APRM AGAFs and thermal limits are in spec. The crew will also determine that the affected APRM has less than 2 LPRM inputs in a level, making the affected APRM inoperable per TS 3.1, Table 3.1.

After Tech Specs have been addressed, the EPR will begin to oscillate, and the CRS will direct actions per PNPS 2.4.37, Turbine Control System Malfunctions. The crew will take control with the MPR, and the EPR power control switch will be placed to off, stabilizing reactor pressure and power. With the EPR removed from service, the plant will enter an administrative LCO requiring both pressure regulators be restored within 2 hours or the plant be < 25% CTP within 4 hours.

After the administrative LCO has been addressed, the 'B' Recirc flow controller will fail upscale resulting in an increase in core flow and reactor power. When the flow controller failure has been diagnosed, the crew will initiate a scoop lockup per PNPS 2.4.20, Reactor Recirculation System Speed or Flow Control System Malfunction. The crew will then take actions per PNPS 2.4.19, Recirculation Pump MG Set Scoop Tube Lockup, including an evaluation of Recirc pump speeds against the Tech Spec 3.6.F limits. The CRS should also identify and brief the crew on the need to trip the 'B' Recirc pump in the event of a reactor SCRAM.

When the required actions have been directed for the Scoop Tube Lockup, the 'D' SRV will indicate open and Torus temperature will rise. The CRS will enter and direct actions per PNPS 2.4.29, Stuck Open Safety Relief Valve; however, the 'D' SRV will fail to close, requiring a manual reactor scram. Following the scram, the RO should trip the 'B' Recirc pump. The CRS will enter and direct EOP-01, and the BOP operator should identify a failure of PASS H2/O2 Sample valve CV91 to isolate, requiring operator action to close the valve. With the SRV still open, Torus cooling will be initiated. The RBCCW inlet to the A RHR HX inlet valve will fail shut requiring additional operator action to lineup RBCCW. When Torus temperature rises to 80°F, EOP-03 will be entered.

After Torus cooling has been placed in service, the 'D' SRV tail pipe will fail, resulting in rising Torus and Drywell pressure, and EOP-03 and EOP-01 will be re-entered on high drywell pressure. Torus and drywell spray will be initiated as Torus bottom pressure continues to rise; however, with the broken SRV tail pipe, Torus bottom pressure will continue to rise, and emergency depressurization will be required prior to exceeding the limits of the Pressure Suppression Pressure curve.

Pilgrim Nuclear Power Station
2007 NRC Initial License Written Examination
Written Examination Outline Methodology

The written examination outline was developed using a proprietary electronic random outline generator developed by Western Technical Services, Inc.

The software was designed to provide a written examination outline in accordance with the criteria contained in NUREG 1021, Revision 9.

The application was developed using Visual Basic code, relying on a true random function based on the PC system clock. The random generator selects topics in a Microsoft Access Database containing Revision 2 of the BWR K&A catalogue. The selected data is then written to a separate data table. The process for selection of topics is similar to the guidance in ES-401, Attachment 1.

The attached outline report and plant specific suppression profile (not used for PILGRIM. Suppressed topics are listed on attached page) report are written directly from the data tables created by the software. Electronic copies of the data tables are on file.

The process used to develop the outlines is as follows:

- For Tier 1 and Tier 2 generic items, only the items required to be included in accordance with ES-401, Attachment 2 are included in the generation process.
- The PILGRIM plant suppression profile lists all suppressed topics, either at the Topic level (System/EPE) or at the statement level. These items were suppressed prior to the electronic generation process.
- Outline is generated for all topics with KA importance ≥2.5.
- 25 SRO topics are randomly selected from Tier 1 AA2 and required generic items, Tier 2 A2 and required generic items, and Tier 3 generic items (All with ties to 10CFR55.43). 75 RO topics are randomly selected to complete the outline, 100 topics total.
- The exam report generated lists the topic (Question) number in the far right column. RO topics are numbered 1-75, and SRO topics are numbered 76–100. The SRO topics are written in red ink for ease of identification.
- Items that are rejected after the initial generation process are automatically placed on the rejected items page. The software tracks whether items are added manually or by random generation, and a report of outline modification may be generated.
- Disposition of any item randomly selected but not included in the outline is documented and included.

Pilgrim Nuclear Power Station
2007 NRC Initial License Written Examination
Written Examination Outline Methodology

The following topics were suppressed because they either do not exist or the function is not performed at Pilgrim:

Emergency and Abnormal Plant Evolutions - Tier 1/Group 1 (RO / SRO)

295027 High Containment Temperature / 5

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2 (RO / SRO)

295011 High Containment Temp / 5

Plant Systems - Tier 2/Group 1 (RO / SRO)

207000 Isolation (Emergency) Condenser
209002 HPCS

Plant Systems - Tier 2/Group 2 (RO / SRO)

201004 RSCS
239003 MSIV Leakage Control