12/27/99

NOTE TO: NRC DOCUMENT CONTROL DESK
MAIL STOP 0-5-D-24

FROM: Virgil Conley, LICENSING ASSISTANT
~~OPERATING LICENSING~~ BRANCH _ REGION I

SUBJECT: OPERATOR LICENSING EXAMINATION ADMINISTERED ON May 7, 10-13, 1999, AT Susquehanna Units 1 + 2
DOCKET NOS 50-387 + 388

ON May 7, 10-13, 1999 OPERATOR LICENSING EXAMINATIONS WERE ADMINISTERED AT THE REFERENCED FACILITY. ATTACHED YOU WILL FIND THE FOLLOWING INFORMATION FOR PROCESSING THROUGH NUDOCS AND DISTRIBUTION TO THE NRC STAFF, INCLUDING THE NRC PDR.

Item #1 a) FACILITY SUBMITTED OUTLINE AND INITIAL EXAM SUBMITTAL DESIGNATED FOR DISTRIBUTION UNDER RIDS CODE A070.
(Preliminary submittal) (+ Final written Outline)

b) AS GIVEN OPERATING EXAMINATION, DESIGNATED FOR DISTRIBUTION UNDER RIDS CODE A070.

Item #2 EXAMINATION REPORT WITH THE AS GIVEN WRITTEN EXAMINATION ATTACHED, DESIGNATED FOR DISTRIBUTION UNDER RIDS CODE IE42.

A070

Initial outline rec. 3/16/99

March 10, 1999

Dear Larry,

Enclosed find the outline for the May 1999 exam for Susquehanna S. E. S. I have also included 20 randomly selected questions from the written exam for your review.

We are in the process of validating the exam parts, so there may be some changes before the final version is completed.

If you have any questions or comments, feel free to contact me.

Sincerely,

Russell B DeVore

Russell B. DeVore

A070

Facility	Susquehanna	Date of Exam: 05/10/99								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	5	5	5				3	5			3	26
	2	3	3	2				4	3			2	17
	Tier Totals	8	8	7				7	8			5	43
2. Plant Systems	1	2	2	2	1	3	2	3	2	2	2	2	23
	2	1		2	2	1	1	1	2	1	1	1	13
	3				1		1		1		1		4
	Tier Totals	3	2	4	4	4	4	4	5	3	4	3	40
3. Generic Knowledge and Abilities					Cat 1		Cat 2		Cat 3		Cat 4		
					5		3		4		5		17

Note:

- Attempt to distribute topics among all K/A Categories: select at least one topic from every K/A category within each tier.
- Actual point totals must match those specified in the table.
- 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.
- Systems/evolutions within each group are identified on the associated outline.
- The shaded areas are not applicable to the category/tier.

Emergency and Abnormal Plant Evolutions - Tier 1/Group 1

Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Pts.
295003	Partial or Complete Loss of A.C. Power				X			AA1.03 Systems necessary to assure safe plant shutdown	4.4	1
295003	Partial or Complete Loss of A.C. Power		X					AK2.04 A.C. electrical loads	3.5	1
295006	SCRAM	X						AK1.02 Shutdown margin	3.7	1
295007	High Reactor Pressure				X			AA1.04 Safety/relief valve operation: Plant-Specific	4.1	1
295009	Low Reactor Water Level						X	2.4.12 Knowledge of general operating crew responsibilities during emergency operations.	3.9	1
295009	Low Reactor Water Level	X						AK1.05 Natural circulation	3.4	1
295010	High Drywell Pressure	X						AK1.01 Downcomer submergence: Mark-I&II	3.4	1
295010	High Drywell Pressure		X					AK2.02 Drywell/suppression chamber differential pressure: Mark-I&II	3.5	1
295013	High Suppression Pool Temperature			X				AK3.02 Limiting heat additions	3.8	1
295013	High Suppression Pool Temperature				X			AA1.02 Systems that add heat to the suppression pool	3.9	1
295014	Inadvertent Reactivity Addition		X					AK2.02 Fuel thermal limits	4.2	1
295015	Incomplete SCRAM					X		AA2.01 Reactor power	4.3	1
295016	Control Room Abandonment					X		AA2.03 Reactor pressure	4.4	1
295016	Control Room Abandonment		X					AK2.01 Remote shutdown panel: Plant-Specific	4.5	1
295017	High Off-Site Release Rate	X						AK1.02 Protection of the general public	4.3	1
295023	Refueling Accidents		X					AK2.02 Fuel pool cooling and cleanup system	3.2	1
295024	High Drywell Pressure					X		EA2.04 Suppression chamber pressure: Plant-Specific	3.9	1
295025	High Reactor Pressure	X						EK1.03 Safety/relief valve tailpipe temperature/pressure relationships	3.8	1
295026	Suppression Pool High Water Temperature						X	2.4.1 Knowledge of EOP entry conditions and immediate action steps.	4.6	1
295026	Suppression Pool High Water Temperature					X		EA2.01 Suppression pool water temperature	4.2	1
295027	High Containment Temperature (Mark III Containment Only)									
295030	Low Suppression Pool Water Level					X		EA2.01 Suppression pool level	4.2	1
295030	Low Suppression Pool Water Level			X				EK3.03 RCIC operation: Plant-Specific	3.7	1
295031	Reactor Low Water Level			X				EK3.04 Steam cooling	4.3	1
295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown						X	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.4	1
295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown			X				EK3.03 Lowering reactor water level	4.5	1
295038	High Off-Site Release Rate			X				EK3.02 System isolations	4.2	1
500000	High Containment Hydrogen Concentration									
	K/A Category Point Totals:	5	5	5	3	5	3	Group Point Total:		26

BWR SRO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2

Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Pts.
295001	Partial or Complete Loss of Forced Core Flow Circulation	X						AK1.02 Power/flow distribution	3.5	1
295001	Partial or Complete Loss of Forced Core Flow Circulation				X			AA1.01 Recirculation system	3.6	1
295002	Loss of Main Condenser Vacuum		X					AK2.05 Feedwater system	2.7	1
295004	Partial or Complete Loss of D.C. Power				X			AA1.02 Systems necessary to assure safe plant shutdown	4.1	1
295005	Main Turbine Generator Trip						X	2.4.48 Ability to interpret control room indications to verify the status and operation of system, and understand how operator action s and directives affect plant and system conditions.	3.8	1
295008	High Reactor Water Level									
295011	High Containment Temperature (Mark III Containment Only)									
295012	High Drywell Temperature		X					AK2.02 Drywell cooling	3.7	1
295018	Partial or Complete Loss of Component Cooling Water		X					AK2.02 Plant operations	3.6	1
295019	Partial or Complete Loss of Instrument Air					X		AA2.02 Status of safety-related instrument air system loads (see AK2.1 - AK2.19)	3.7	1
295020	Inadvertent Containment Isolation			X				AK3.04 Reactor pressure response	4.1	1
295021	Loss of Shutdown Cooling				X			AA1.04 Alternate heat removal methods	3.7	1
295022	Loss of CRD Pumps	X						AK1.01 Reactor pressure vs. rod insertion capability	3.4	1
295028	High Drywell Temperature	X						EK1.01 Reactor water level measurement	3.7	1
295029	High Suppression Pool Water Level					X		EA2.03 Drywell/containment water level	3.5	1
295032	High Secondary Containment Area Temperature						X	2.2.3 (multi-unit) Knowledge of the design, procedural, and operational differences between units.	3.3	1
295033	High Secondary Containment Area Radiation Levels				X			EA1.03 Secondary containment ventilation	3.8	1
295034	Secondary Containment Ventilation High Radiation									
295035	Secondary Containment High Differential Pressure									
295036	Secondary Containment High Sump/Area Water Level					X		EA2.02 Water level in the affected area	3.1	1
600000	Plant Fire On Site			X				EK3.04 Actions contained in the abnormal procedure for plant fire on site	3.4	1
	K/A Category Point Totals:	3	3	2	4	3	2	Group Point Total:		17

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Pts.
201005	Rod Control and Information System (RCIS)														
202002	Recirculation Flow Control System							X					A1.01 Recirculation pump speed: BWR-2, 3, 4, 5, 6	3.2	1
203000	RHR/LPCI: Injection Mode (Plant Specific)		X										K2.03 Initiation logic	2.9	1
206000	High Pressure Coolant Injection System					X							K5.05 Turbine speed control: BWR-2, 3, 4	3.3	1
206000	High Pressure Coolant Injection System						X						K6.09 Condensate storage and transfer system: BWR-2, 3, 4	3.5	1
207000	Isolation (Emergency) Condenser														
209001	Low Pressure Core Spray System	X											K1.01 Condensate storage tank: Plant-Specific	3.1	1
209002	High Pressure Core Spray System (HPCS)														
211000	Standby Liquid Control System			X									K3.01 Ability to shutdown the reactor in certain conditions	4.4	1
211000	Standby Liquid Control System								X				A2.01 Pump trip	3.8	1
212000	Reactor Protection System											X	2.1.12 Ability to apply technical specifications for a system.	4.0	1
212000	Reactor Protection System								X				A2.19 Partial system activation (half-SCRAM)	3.9	1
215004	Source Range Monitor (SRM) System											X	2.2.26 Knowledge of refueling administrative requirements.	3.7	1
215005	Average Power Range Monitor/Local Power Range Monitor System							X					A1.07 APRM (gain adjustment factor)	3.4	1
216000	Nuclear Boiler Instrumentation			X									K3.24 Vessel level monitoring	4.1	1
217000	Reactor Core Isolation Cooling System (RCIC)										X		A4.01 RCIC turbine speed	3.7	1
217000	Reactor Core Isolation Cooling System (RCIC)									X			A3.01 Valve operation	3.5	1
218000	Automatic Depressurization System		X										K2.01 ADS logic	3.3	1
218000	Automatic Depressurization System					X							K5.01 ADS logic operation	3.8	1
223001	Primary Containment System and Auxiliaries														
223002	Primary Containment Isolation System/Nuclear Steam Supply Shut-Off				X								K4.05 Single failures will not impair the function ability of the system	3.1	1
226001	RHR/LPCI: Containment Spray System Mode														
239002	Relief/Safety Valves					X							K5.06 Vacuum breaker operation	3.0	1

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

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Pts.
241000	Reactor/Turbine Pressure Regulating System						X						K6.20 Main generator	3.0	1
259002	Reactor Water Level Control System										X		A4.10 Setpoint setdown reset controls: Plant-Specific	2.9	1
261000	Standby Gas Treatment System	X											K1.06 High pressure coolant injection system: Plant- Specific	3.1	1
262001	A.C. Electrical Distribution							X					A1.05 Breaker lineups	3.5	1
264000	Emergency Generators (Diesel/Jet)									X			A3.01 Automatic starting of compressor and emergency generator	3.1	1
290001	Secondary Containment														
	K/A Category Point Totals:	2	2	2	1	3	2	3	2	2	2	2	Group Point Total:		23

Plant Systems - Tier 2/Group 2

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Pts.
201001	Control Rod Drive Hydraulic System				X								K4.04 Scramming control rods with inoperative SCRAM solenoid valves (back-up SCRAM valves)	3.6	1
201002	Reactor Manual Control System	X											K1.01 Control rod drive hydraulic system	3.2	1
201004	Rod Sequence Control System (Plant Specific)														
201006	Rod Worth Minimizer System (RWM) (Plant Specific)								X				A2.05 Out of sequence rod movement; P-Spec(Not-BWR6)	3.5	1
202001	Recirculation System								X				A2.08 Recirculation flow mismatch: Plant-Specific	3.4	1
204000	Reactor Water Cleanup System									X			A3.03 Response to system isolations	3.6	1
205000	Shutdown Cooling System (RHR Shutdown Cooling Mode)			X									K3.02 Reactor water level: Plant-Specific	3.3	1
214000	Rod Position Information System														
215002	Rod Block Monitor System						X						K6.05 LPRM detectors: BWR-3, 4, 5	3.1	1
215003	Intermediate Range Monitor (IRM) System										X		A4.07 Verification of proper functioning/ operability	3.6	1
219000	RHR/LPCI: Torus/Suppression Pool Cooling Mode														
230000	RHR/LPCI: Torus/Suppression Pool Spray Mode														
234000	Fuel Handling Equipment					X							K5.02 Fuel handling equipment interlocks	3.7	1
239003	MSIV Leakage Control System														
245000	Main Turbine Generator and Auxiliary Systems							X					A1.07 First stage turbine pressure	2.8	1
259001	Reactor Feedwater System														
262002	Uninterruptable Power Supply (A.C./D.C.)														
263000	D.C. Electrical Distribution											X	2.1.12 Ability to apply technical specifications for a system.	4.0	1
271000	Offgas System														
272000	Radiation Monitoring System				X								K4.01 Redundancy	2.8	1
286000	Fire Protection System			X									K3.03 Plant protection	3.8	1
290003	Control Room HVAC														
300000	Instrument Air System (IAS)														
400000	Component Cooling Water System (CCWS)														
	K/A Category Point Totals:	1	0	2	2	1	1	1	2	1	1	1	Group Point Total:		13

ES-401		BWR SRO Examination Outline Plant Systems - Tier 2/Group 3													ES-401-1
Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Pts.
201003	Control Rod and Drive Mechanism										X		A4.02 CRD mechanism position: Plant-Specific	3.5	1
215001	Traversing In-Core Probe				X								K4.01 Primary containment isolation: Mark-I&II(Not-BWR1)	3.5	1
233000	Fuel Pool Cooling and Clean-up														
239001	Main and Reheat Steam System						X						K6.02 Plant air systems	3.2	1
256000	Reactor Condensate System														
268000	Radwaste														
288000	Plant Ventilation Systems														
290002	Reactor Vessel Internals								X				A2.05 Exceeding thermal limits	4.2	1
	K/A Category Point Totals:	0	0	0	1	0	1	0	1	0	1	0	Group Point Total:		4

Facility Susquehanna	Date:	May 10, 1999	Exam Level:	SRO
Category	KA #	KA Topic	Imp.	Points
Conduct of Operations	2.1.9	Ability to direct personnel activities inside the control room.	4.0	1
	2.1.12	Ability to apply technical specifications for a system.	4.0	1
	2.1.12	Ability to apply technical specifications for a system.	4.0	1
	2.1.14	Knowledge of system status criteria which require the notification of plant personnel.	3.3	1
	2.1.21	Ability to obtain and verify controlled procedure copy.	3.2	1
	Total			5
Equipment Control	2.2.13	Knowledge of tagging and clearance procedures.	3.8	1
	2.2.14	Knowledge of the process for making configuration changes.	3.0	1
	2.2.23	Ability to track limiting conditions for operations.	3.8	1
	Total			3
Radiation Control	2.3.1	Knowledge of 10 CFR 20 and related facility radiation control requirements.	3.0	1
	2.3.2	Knowledge of facility ALARA program.	2.9	1
	2.3.4	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	3.1	1
	2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	3.3	1
	Total			4
Emergency Procedures and Plan	2.4.25	Knowledge of fire protection procedures.	3.4	1
	2.4.38	Ability to take actions called for in the facility emergency plan, including (if required)supporting or acting as emergency coordinator.	4.0	1
	2.4.40	Knowledge of the SRO's responsibilities in emergency plan implementation.	4.0	1
	2.4.41	Knowledge of the emergency action level thresholds and classifications.	4.1	1
	2.4.49	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.0	1
	Total			5
Tier 3 Target Point Total (RO/SRO)				17

Facility: Susquehanna **Examination Level:** SRO		**Date of Examination:** 05/10/99 **Operating Test Number:**
Administrative Topic/Subject Description		**Describe method of evaluation:** **1. ONE Administrative JPM, OR** **2. TWO Administrative Questions**
A.1	Plant Cooldown Limits	JPM - Calculate a cooldown rate in accordance with SO-100-011
	Reactor Startup Requirements	JPM - Determine if Rod Worth Minimizer bypassing can be authorized in accordance with NDAP-QA-0338
A.2	Technical Specification 3.0.3 actions	JPM - Determine plant equipment operability and implement Tech Spec 3.0.3 including documentation/reports
A.3	Liquid Radioactive Waste Releases	Ques #1- Liquid Radwaste Rad Monitor operation during releases
		Ques #2- Cooling Tower blowdown flow during releases
A.4	Emergency Director Actions	JPM - Classify and make protective action recommendations for a General Emergency

Initial outline

Facility: Susquehanna **Examination Level:** SRO(I)		**Date of Examination:** 05/10/99 **Operating Test No:**
System / JPM Title / Type Codes*	**Safety Function**	**Planned Follow-up Questions: K/A/G - Importance - Description**
1. RMCS/Take Actions For A Control Rod Double Notch/S,N	1	a. 201001K407 - 3.1/3.0 - PCV and FCV operational relationship b. 201001K110 - 2.8/2.8 - Control rod speed adjustments
2. HPCI/Place HPCI In CST To CST Mode - Steam Leak w/o Isolation/S,N,A	2	a. 206000K505 - 3.3/3.3 - Failed ramp generator b. 206000A217 - 3.9/4.3 - HPCI operation vs loss of feedwater heating
3. RHR-LPCI/Transfer From SDC To LPCI On Low Water Level/S,L,M	4	a. 295021K201 - 3.6/3.7 - Mode change/Tech Specs b. 205000G222 - 3.4/4.1 - Inop pressure switch - SDC Iso
4. RHR-SPC Mode/Place Suppression Pool Cooling In Service From RSDP/S,D	5	a. 219000A201 - 3.0/3.1 - RHR Pump vortex limits b. 295016K201 - 4.4/4.5 - RHR operations at RSDP
5. Diesel Gen/Synch "B" DG To 4.16KV Bus 2B-Runaway Diesel/S,M,A	6	a. 264000A210 - 3.9/4.2 - DG response to LOCA while paralleled b. 264000A404 - 3.7/3.7 - DG emergency stop actions and response
6. RSCS/Bypass Control Rod In Rod Sequence Control System/S,M	7	a. 201004K301 - 3.3/3.4 - Rod movement with Inop RSCS b. 201004K501 - 3.6/3.0 - Fuel damage magnitude during rod drop accident
7. SGT/Perform Manual Startup Of SGTS & Vent The Drywell/S,M	9	a. 261000G112 - 2.9/4.0 - Inop SGTS vs Secondary Containment Integrity b. 261000A103 - 3.2/3.8 - Off-site doses at site boundary
8. SLC/Connect SLC Storage Tank To RCIC/D,P,R	1	a. 211000A205 - 3.1/3.4 - T.S. actions for SLC pump suction temperatures b. 211000A109 - 4.0/4.1 - Local pump operations
9. PCIS/Bypass MSIV And MSL Drain Isolation Signals/D,P	5	a. 223002K408 - 3.3/3.7 - One jumper not installed affect on isolation logic. b. 295037K306 - 3/8/4.1 - ES requirements vs plant conditions
10. Fire Prot/Fire Protection Crosstie To RHRSW/M,P	8	a. 286000A406 - 3.4/3.4 - Diesel Fire Pump failure to start b. 295031A108 - 3.8/3.9 - Minimum fire water available

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

#6 moved to #10
NEW #6 written.
#10 Gone
in orig. exam submittal.

File Name: JPM Out-SROI Date and Time Printed: 03/10/99 11:09 AM

Facility: Susquehanna **Examination Level:** SRO(U)		**Date of Examination:** 05/10/99 **Operating Test No:**
System / JPM Title / Type Codes*	**Safety Function**	**Planned Follow-up Questions: K/A/G - Importance - Description**
1. N/A		a.
		b.
2. N/A		a.
		b.
3. RHR-LPCI/Transfer From SDC To LPCI	4	a. 295021K201 - 3.6/3.7 - Mode change/Tech Specs
On Low Water Level/S,L,M		b. 205000G222 - 3.4/4.1 - Inop pressure switch - SDC Iso
4. RHR-SPC Mode/Place Suppression Pool	5	a. 219000A201 - 3.0/3.1 - RHR Pump vortex limits
Cooling In Service From RSDP/S,D		b. 295016K201 - 4.4/4.5 - RHR operations at RSDP
5. Diesel Gen/Synch "B" DG To 4.16KV	6	a. 264000A210 - 3.9/4.2 - DG response to LOCA while paralleled
Bus 2B-Runaway Diesel/S,M,A		b. 264000A404 - 3.7/3.7 - DG emergency stop actions and response
6. N/A		a.
		b.
7. N/A		a.
		b.
8. SLC/Connect SLC Storage Tank To	1	a. 211000A205 - 3.1/3.4 - T.S. actions for SLC pump suction temperatures
RCIC/D,P,R		b. 211000A109 - 4.0/4.1 - Local pump operations
9. N/A		a.
		b.
10. Fire Prot/Fire Protection Crosstie To	8	a. 286000A406 - 3.4/3.4 - Diesel Fire Pump failure to start
RHRSW/M,P		b. 295031A108 - 3.8/3.9 - Minimum fire water available

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

Facility: Susquehanna **Examination Level:** SRO		**Date of Examination:** 05/10/99 **Operating Test Number:**
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A.2	Technical Specification 3.0.3 actions	JPM - Determine plant equipment operability and implement Tech Spec 3.0.3 including documentation/reports
A.3	Liquid Radioactive Waste Releases	Ques - Liquid Radwaste Rad Monitor operation during releases
		Ques - Cooling Tower blowdown flow during releases
A.4	Emergency Director Actions	JPM - Classify and make protective action recommendations for a General Emergency

(TAGS)

SUSQUEHANNA

NRC WRITTEN EXAM

OUTLINES

Facility	Susquehanna	Date of Exam: 05/10/99								Exam Level:	SRO	
Tier	Group	K/A Category Points										Point Total
		K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	
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	3				1		1		1		1		4
	Tier Totals	3	2	4	4	4	4	4	5	3	4	3	40

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

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- Actual point totals must match those specified in the table.
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- Systems/evolutions within each group are identified on the associated outline.
- The shaded areas are not applicable to the category/tier.

Required
26
17
43
23
13
4
40
17

Emergency and Abnormal Plant Evolutions - Tier 1/Group 1

Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Pts.
295003	Partial or Complete Loss of A.C. Power				X			AA1.03 Systems necessary to assure safe plant shutdown	4.4	1
295003	Partial or Complete Loss of A.C. Power		X					AK2.04 A.C. electrical loads	3.5	1
295006	SCRAM	X						AK1.02 Shutdown margin	3.7	1
295007	High Reactor Pressure				X			AA1.04 Safety/relief valve operation: Plant-Specific	4.1	1
295009	Low Reactor Water Level						X	2.4.12 Knowledge of general operating crew responsibilities during emergency operations.	3.9	1
295009	Low Reactor Water Level	X						AK1.05 Natural circulation	3.4	1
295010	High Drywell Pressure	X						AK1.01 Downcomer submergence: Mark-I&II	3.4	1
295010	High Drywell Pressure		X					AK2.02 Drywell/suppression chamber differential pressure: Mark-I&II	3.5	1
295013	High Suppression Pool Temperature			X				AK3.02 Limiting heat additions	3.6	1
295013	High Suppression Pool Temperature				X			AA1.02 Systems that add heat to the suppression pool	3.9	1
295014	Inadvertent Reactivity Addition		X					AK2.02 Fuel thermal limits	4.2	1
295015	Incomplete SCRAM					X		AA2.01 Reactor power	4.3	1
295016	Control Room Abandonment					X		AA2.03 Reactor pressure	4.4	1
295016	Control Room Abandonment		X					AK2.01 Remote shutdown panel: Plant-Specific	4.5	1
295017	High Off-Site Release Rate	X						AK1.02 Protection of the general public	4.3	1
295023	Refueling Accidents		X					AK2.02 Fuel pool cooling and cleanup system	3.2	1
295024	High Drywell Pressure					X		EA2.04 Suppression chamber pressure: Plant-Specific	3.9	1
295025	High Reactor Pressure	X						EK1.03 Safety/relief valve tailpipe temperature/pressure relationships	3.6	1
295026	Suppression Pool High Water Temperature						X	2.4.1 Knowledge of EOP entry conditions and immediate action steps.	4.6	1
295026	Suppression Pool High Water Temperature					X		EA2.01 Suppression pool water temperature	4.2	1
295027	High Containment Temperature (Mark III Containment Only)									
295030	Low Suppression Pool Water Level					X		EA2.01 Suppression pool level	4.2	1
295030	Low Suppression Pool Water Level			X				EK3.03 RCIC operation: Plant-Specific	3.7	1
295031	Reactor Low Water Level			X				EK3.04 Steam cooling	4.3	1
295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown						X	2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	4.4	1
295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown			X				EK3.03 Lowering reactor water level	4.5	1
295038	High Off-Site Release Rate			X				EK3.02 System isolations	4.2	1
500000	High Containment Hydrogen Concentration									
	K/A Category Point Totals:	5	5	5	3	5	3	Group Point Total:		26

BWR SRO Examination Outline

Emergency and Abnormal Plant Evolutions - Tier 1/Group 2

Number#	Name	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Pts.
295001	Partial or Complete Loss of Forced Core Flow Circulation	X						AK1.02 Power/flow distribution	3.5	1
295001	Partial or Complete Loss of Forced Core Flow Circulation				X			AA1.01 Recirculation system	3.6	1
295002	Loss of Main Condenser Vacuum		X					AK2.05 Feedwater system	2.7	1
295004	Partial or Complete Loss of D.C. Power				X			AA1.02 Systems necessary to assure safe plant shutdown	4.1	1
295005	Main Turbine Generator Trip						X	2.4.48 Ability to interpret control room indications to verify the status and operation of system, and understand how operator action s and directives affect plant and system conditions.	3.8	1
295008	High Reactor Water Level									
295011	High Containment Temperature (Mark III Containment Only)									
295012	High Drywell Temperature		X					AK2.02 Drywell cooling	3.7	1
295018	Partial or Complete Loss of Component Cooling Water		X					AK2.02 Plant operations	3.6	1
295019	Partial or Complete Loss of Instrument Air					X		AA2.02 Status of safety-related instrument air system loads (see AK2.1 - AK2.19)	3.7	1
295020	Inadvertent Containment Isolation			X				AK3.04 Reactor pressure response	4.1	1
295021	Loss of Shutdown Cooling				X			AA1.04 Alternate heat removal methods	3.7	1
295022	Loss of CRD Pumps	X						AK1.01 Reactor pressure vs. rod insertion capability	3.4	1
295028	High Drywell Temperature	X						EK1.01 Reactor water level measurement	3.7	1
295029	High Suppression Pool Water Level					X		EA2.03 Drywell/containment water level	3.5	1
295032	High Secondary Containment Area Temperature						X	2.2.3 (multi-unit) Knowledge of the design, procedural, and operational differences between units.	3.3	1
295033	High Secondary Containment Area Radiation Levels				X			EA1.03 Secondary containment ventilation	3.8	1
295034	Secondary Containment Ventilation High Radiation									
295035	Secondary Containment High Differential Pressure									
295036	Secondary Containment High Sump/Area Water Level					X		EA2.02 Water level in the affected area	3.1	1
600000	Plant Fire On Site			X				EK3.04 Actions contained in the abnormal procedure for plant fire on site	3.4	1
	K/A Category Point Totals:	3	3	2	4	3	2	Group Point Total:		17

Plant Systems - Tier 2/Group 1

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Pts.
201005	Rod Control and Information System (RCIS)														
202002	Recirculation Flow Control System							X					A1.01 Recirculation pump speed: BWR-2, 3, 4, 5, 6	3.2	1
203000	RHR/LPCI: Injection Mode (Plant Specific)		X										K2.03 Initiation logic	2.9	1
206000	High Pressure Coolant Injection System					X							K5.05 Turbine speed control: BWR-2, 3, 4	3.3	1
206000	High Pressure Coolant Injection System						X						K6.09 Condensate storage and transfer system: BWR-2, 3, 4	3.5	1
207000	Isolation (Emergency) Condenser														
209001	Low Pressure Core Spray System	X											K1.01 Condensate storage tank: Plant-Specific	3.1	1
209002	High Pressure Core Spray System (HPCS)														
211000	Standby Liquid Control System			X									K3.01 Ability to shutdown the reactor in certain conditions	4.4	1
211000	Standby Liquid Control System								X				A2.01 Pump trip	3.8	1
212000	Reactor Protection System											X	2.1.12 Ability to apply technical specifications for a system.	4.0	1
212000	Reactor Protection System								X				A2.19 Partial system activation (half-SCRAM)	3.9	1
215004	Source Range Monitor (SRM) System											X	2.2.26 Knowledge of refueling administrative requirements.	3.7	1
215005	Average Power Range Monitor/Local Power Range Monitor System							X					A1.07 APRM (gain adjustment factor)	3.4	1
216000	Nuclear Boiler Instrumentation			X									K3.24 Vessel level monitoring	4.1	1
217000	Reactor Core Isolation Cooling System (RCIC)										X		A4.01 RCIC turbine speed	3.7	1
217000	Reactor Core Isolation Cooling System (RCIC)									X			A3.01 Valve operation	3.5	1
218000	Automatic Depressurization System		X										K2.01 ADS logic	3.3	1
218000	Automatic Depressurization System					X							K5.01 ADS logic operation	3.8	1
223001	Primary Containment System and Auxiliaries														
223002	Primary Containment Isolation System/Nuclear Steam Supply Shut-Off				X								K4.05 Single failures will not impair the function ability of the system	3.1	1
226001	RHR/LPCI: Containment Spray System Mode														
239002	Relief/Safety Valves					X							K5.06 Vacuum breaker operation	3.0	1

BWR SRO Examination Outline

Plant Systems - Tier 2/Group 1

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Pts.
241000	Reactor/Turbine Pressure Regulating System						X						K6.20 Main generator	3.0	1
259002	Reactor Water Level Control System										X		A4.10 Setpoint seldown reset controls: Plant-Specific	2.9	1
261000	Standby Gas Treatment System	X											K1.06 High pressure coolant injection system: Plant- Specific	3.1	1
262001	A.C. Electrical Distribution							X					A1.05 Breaker lineups	3.5	1
264000	Emergency Generators (Diesel/Jet)									X			A3.01 Automatic starting of compressor and emergency generator	3.1	1
290001	Secondary Containment														
	K/A Category Point Totals:	2	2	2	1	3	2	3	2	2	2	2	Group Point Total:		23

Plant Systems - Tier 2/Group 2

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Pts.
201001	Control Rod Drive Hydraulic System				X								K4.04 Scramming control rods with inoperative SCRAM solenoid valves (back-up SCRAM valves)	3.6	1
201002	Reactor Manual Control System	X											K1.01 Control rod drive hydraulic system	3.2	1
201004	Rod Sequence Control System (Plant Specific)														
201006	Rod Worth Minimizer System (RWM) (Plant Specific)								X				A2.05 Out of sequence rod movement; P-Spec(Not-BWR6)	3.5	1
202001	Recirculation System								X				A2.08 Recirculation flow mismatch: Plant-Specific	3.4	1
204000	Reactor Water Cleanup System									X			A3.03 Response to system isolations	3.6	1
205000	Shutdown Cooling System (RHR Shutdown Cooling Mode)			X									K3.02 Reactor water level: Plant-Specific	3.3	1
214000	Rod Position Information System														
215002	Rod Block Monitor System						X						K6.05 LPRM detectors: BWR-3, 4, 5	3.1	1
215003	Intermediate Range Monitor (IRM) System										X		A4.07 Verification of proper functioning/ operability	3.6	1
219000	RHR/LPCI: Torus/Suppression Pool Cooling Mode														
230000	RHR/LPCI: Torus/Suppression Pool Spray Mode														
234000	Fuel Handling Equipment					X							K5.02 Fuel handling equipment interlocks	3.7	1
239003	MSIV Leakage Control System														
245000	Main Turbine Generator and Auxiliary Systems							X					A1.07 First stage turbine pressure	2.8	1
259001	Reactor Feedwater System														
262002	Uninterruptable Power Supply (A.C./D.C.)														
263000	D.C. Electrical Distribution											X	2.1.12 Ability to apply technical specifications for a system.	4.0	1
271000	Offgas System														
272000	Radiation Monitoring System				X								K4.01 Redundancy	2.8	1
286000	Fire Protection System			X									K3.03 Plant protection	3.6	1
290003	Control Room HVAC														
300000	Instrument Air System (IAS)														
400000	Component Cooling Water System (CCWS)														
	K/A Category Point Totals:	1	0	2	2	1	1	1	2	1	1	1	Group Point Total:		13

Plant Systems - Tier 2/Group 3

Number#	Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Pts.
201003	Control Rod and Drive Mechanism										X		A4.02 CRD mechanism position: Plant-Specific	3.5	1
215001	Traversing In-Core Probe				X								K4.01 Primary containment isolation: Mark-I&II(Not-BWR1)	3.5	1
233000	Fuel Pool Cooling and Clean-up														
239001	Main and Reheat Steam System						X						K6.02 Plant air systems	3.2	1
256000	Reactor Condensate System														
268000	Radwaste														
288000	Plant Ventilation Systems														
290002	Reactor Vessel Internals								X				A2.05 Exceeding thermal limits	4.2	1
	K/A Category Point Totals:	0	0	0	1	0	1	0	1	0	1	0	Group Point Total:		4

Facility Susquehanna	Date: May 10, 1999		Exam Level:	SRO
Category	KA #	KA Topic	Imp.	Points
Conduct of Operations	2.1.9	Ability to direct personnel activities inside the control room.	4.0	1
	2.1.12	Ability to apply technical specifications for a system.	4.0	1
	2.1.12	Ability to apply technical specifications for a system.	4.0	1
	2.1.14	Knowledge of system status criteria which require the notification of plant personnel.	3.3	1
	2.1.21	Ability to obtain and verify controlled procedure copy.	3.2	1
	Total			5
Equipment Control	2.2.13	Knowledge of tagging and clearance procedures.	3.8	1
	2.2.14	Knowledge of the process for making configuration changes.	3.0	1
	2.2.23	Ability to track limiting conditions for operations.	3.8	1
	Total			3
Radiation Control	2.3.1	Knowledge of 10 CFR 20 and related facility radiation control requirements.	3.0	1
	2.3.2	Knowledge of facility ALARA program.	2.9	1
	2.3.4	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	3.1	1
	2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	3.3	1
	Total			4
Emergency Procedures	2.4.25	Knowledge of fire protection procedures.	3.4	1
and Plan	2.4.38	Ability to take actions called for in the facility emergency plan, including (if required)supporting or acting as emergency coordinator.	4.0	1
	2.4.40	Knowledge of the SRO's responsibilities in emergency plan implementation.	4.0	1
	2.4.41	Knowledge of the emergency action level thresholds and classifications.	4.1	1
	2.4.49	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	4.0	1
	Total			5
Tier 3 Target Point Total (RO/SRO)				17

SUSQUEHANNA

NRC WRITTEN EXAM

OUTLINES

W/ QUESTION TOPIC

Facility: Susquehanna | Exam Date: 5/10/99 | Examination Level: SRO

Section Title Generic Knowledge and Abilities

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Lev	Question Topic
GENERIC	2.1.9	4.0	Ability to direct personnel activities inside the control room.	S	Specific Unit Supervisor responsibilities
	2.1.12	4.0	Ability to apply technical specifications for a system.	S	LCO 3.0.3 actions
	2.1.12	4.0	Ability to apply technical specifications for a system.	S	Tech Spec completion times
	2.1.14	3.3	Knowledge of system status criteria which require the notification of plant personnel.	S	Reactor Mode Switch to "Startup" approval
	2.1.21	3.2	Ability to obtain and verify controlled procedure copy.	S	User Controlled copies limited to 24 hour use
	2.2.13	3.8	Knowledge of tagging and clearance procedures.	S	Verifying positions of inaccessible valves
	2.2.14	3.0	Knowledge of the process for making configuration changes.	S	Tracking Checkoff List Status Changes
	2.2.23	3.8	Ability to track limiting conditions for operations.	S	Maximum Out Of Service Time calculation and application
	2.3.1	3.0	Knowledge of 10 CFR 20 and related facility radiation control requirements.	S	Entry into an HP Controlled Area from the RCA
	2.3.2	2.9	Knowledge of facility ALARA program.	S	ALARA considerations for exposure
	2.3.4	3.1	Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.	S	Exposure extensions during a declared emergency
	2.3.10	3.3	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	S	Red tagging components in a High Radiation Area
	2.4.25	3.4	Knowledge of fire protection procedures.	S	Firewatch tours in High Radiation Areas
	2.4.38	4.0	Ability to take actions called for in the facility emergency plan, including (if required)supporting or acting as emergency coordinator.	S	10CFR50.54(x) & (Y) criteria
	2.4.40	4.0	Knowledge of the SRO's responsibilities in emergency plan implementation.	S	Time from meeting EAL to classification on a SAE

Facility: Susquehanna   Exam Date: 5/10/99   Examination Level: SRO

Section Title Generic Knowledge and Abilities

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Lev	Question Topic
GENERIC	2.4.41	4.1	Knowledge of the emergency action level thresholds and classifications.	S	Classifications following momentary exceeding of EAL
	2.4.49	4.0	Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	S	Operator actions when an expected automatic action did not occur

Facility: Susquehanna | Exam Date: 5/10/99 | Examination Level: SRO

Section Title Plant Systems

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Lev	Question Topic
202002	A1.01	3.2	Recirculation pump speed: BWR-2, 3, 4, 5, 6	S	Recirc Pump Limiter operations
203000	K2.03	2.9	Initiation logic	S	Loss of power to one division of RHR Initiation logic
206000	K5.05	3.3	Turbine speed control: BWR-2, 3, 4	S	HPCI Ramp Generator failure will operating in automatic
	K6.09	3.5	Condensate storage and transfer system: BWR-2, 3, 4	S	HPCI support equipment vs operable
209001	K1.01	3.1	Condensate storage tank: Plant-Specific	S	Core Spray operability while lined up to the CST
211000	A2.01	3.8	Pump trip	S	Cold Shutdown Boron Injected criteria
	K3.01	4.4	Ability to shutdown the reactor in certain conditions	S	SLC "Subsystem" criteria and operability
212000	A2.19	3.9	Partial system activation (half-SCRAM)	S	RPS vs Backup Scram Valve relationship
	2.1.12	4.0	Ability to apply technical specifications for a system.	S	Actions for Inoperable RPS EPAs
215004	2.2.26	3.7	Knowledge of refueling administrative requirements.	S	SRM operability during fuel load
215005	A1.07	3.4	APRM (gain adjustment factor)	S	APRM Gain Adjustment requirements
216000	K3.24	4.1	Vessel level monitoring	S	Excess flow check valve closure effects on level
217000	A3.01	3.5	Valve operation	S	RCIC suction sources on initiation signal
	A4.01	3.7	RCIC turbine speed	S	RCIC response to loss of oil pressure while operating
218000	K2.01	3.3	ADS logic	S	Loss of power affects on ADS logic
	K5.01	3.8	ADS logic operation	S	Remote Shutdown Panel SRV vs ADS operation
223002	K4.05	3.1	Single failures will not impair the function ability of the system	S	Actions for Inop MSIV
239002	K5.06	3.0	Vacuum breaker operation	S	Failed SRV vacuum breaker indications
241000	K6.20	3.0	Main generator	S	Load reject circuits/Intercept Valve fast closure

Facility: Susquehanna | Exam Date: 5/10/99 | Examination Level: SRO

Section Title Plant Systems

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Lev	Question Topic
259002	A4.10	2.9	Setpoint setdown reset controls: Plant-Specific	S	Setpoint setdown operation
261000	K1.06	3.1	High pressure coolant injection system: Plant-Specific	S	SGTS operation during a LOCA
262001	A1.05	3.5	Breaker lineups	S	ESS Bus transfers
264000	A3.01	3.1	Automatic starting of compressor and emergency generator	S	Local mode operation of the Diesel Generators

Facility: Susquehanna | Exam Date: 5/10/99 | Examination Level: SRO

Section Title Plant Systems

SRO Group 2

System/Evolution	K/A	SRO	KA Statement	Lev	Question Topic
201001	K4.04	3.6	Scramming control rods with inoperative SCRAM solenoid valves (back-up SCRAM valves)	S	"All rods in" times on a Backup Scram Valve initiated scram
201002	K1.01	3.2	Control rod drive hydraulic system	S	Indications of failed open Scram Outlet Valve
201006	A2.05	3.5	Out of sequence rod movement; P-Spec(Not-BWR6)	S	RWM Insert errors
202001	A2.08	3.4	Recirculation flow mismatch: Plant-Specific	S	Mismatched Recirc flow limitations
204000	A3.03	3.6	Response to system isolations	S	Emergency Support Procedure affects on RWCU in Blowdown Mode
205000	K3.02	3.3	Reactor water level: Plant-Specific	S	LPCI Injection Valve operation following SDC isolation
215002	K6.05	3.1	LPRM detectors: BWR-3, 4, 5	S	RBM Gain Change Circuit failure
215003	A4.07	3.6	Verification of proper functioning/ operability	S	IRM Downscale rod blocks
234000	K5.02	3.7	Fuel handling equipment interlocks	S	Use of fuel Hoist Override during core alts.
245000	A1.07	2.8	First stage turbine pressure	S	Exceeding 1st stage pressures during shell warming
263000	2.1.12	4.0	Ability to apply technical specifications for a system.	S	Inop battery Tech Specs
272000	K4.01	2.8	Redundancy	S	MSL Rad Monitor vs MSIV closures
286000	K3.03	3.8	Plant protection	S	SGTS operability vs fire suppression

Facility: Susquehanna | Exam Date: 5/10/99 | Examination Level: SRO

Section Title Plant Systems

SRO Group 3

System/Evolution	K/A	SRO	KA Statement	Lev	Question Topic
201003	A4.02	3.5	CRD mechanism position: Plant-Specific	S	Uncoupled control rod indications
215001	K4.01	3.5	Primary containment isolation: Mark-I&II(Not-BWR1)	S	TIP Panel indications
239001	K6.02	3.2	Plant air systems	S	MSIV operator actuator design
290002	A2.05	4.2	Exceeding thermal limits	S	APRM adjustments with MFLPD greater than RTP

Facility: Susquehanna | Exam Date: 5/10/99 | Examination Level: SRO

Section Title Emergency and Abnormal Plant Evolutions

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Lev	Question Topic
295003	AA1.03	4.4	Systems necessary to assure safe plant shutdown	S	RCIC operation during station blackout
	AK2.04	3.5	A.C. electrical loads	S	Loss of one RPS bus affect on condenser vacuum
295006	AK1.02	3.7	Shutdown margin	S	EO-113 entry from ON-100-101
295007	AA1.04	4.1	Safety/relief valve operation: Plant-Specific	S	ADS operation with TBV already open
295009	2.4.12	3.9	Knowledge of general operating crew responsibilities during emergency operations.	S	RWCU system isolation requirements
	AK1.05	3.4	Natural circulation	S	Reason for reducing CRD flow post scram with no recirc running
295010	AK1.01	3.4	Downcomer submergence: Mark-I&II	S	Drywell and suppression chamber pressure relationship during slow drywell pressure increases
	AK2.02	3.5	Drywell/suppression chamber differential pressure: Mark-I&II	S	Drywell Spray Initiation Limit - RHR Pump flow limited on startup
295013	AA1.02	3.9	Systems that add heat to the suppression pool	S	Indications of a stuck open SRV
	AK3.02	3.8	Limiting heat additions	S	Margin to the HCTL while operating SRVs
295014	AK2.02	4.2	Fuel thermal limits	S	Loss of feedwater heating versus thermal limits
295015	AA2.01	4.3	Reactor power	S	EOP actions on a scram with RPIS Inop
295016	AA2.03	4.4	Reactor pressure	S	Status of Recirc on Control Room Evacuation
	AK2.01	4.5	Remote shutdown panel: Plant-Specific	S	Reactor water level control from the Remote Shutdown Panel
295017	AK1.02	4.3	Protection of the general public	S	Purpose of EO-100-108
295023	AK2.02	3.2	Fuel pool cooling and cleanup system	S	Preventing draining fuel pool to suppression pool during RHR Fuel Pool Cooling Mode
295024	EA2.04	3.9	Suppression chamber pressure: Plant-Specific	S	Indications of a failed SRV tail pipe in the suppression chamber
295025	EK1.03	3.8	Safety/relief valve tailpipe temperature/pressure relationships	S	SRV tailpipe temp trend during depressurization

Facility: Susquehanna　　Exam Date: 5/10/99　　Examination Level: SRO

Section Title Emergency and Abnormal Plant Evolutions

SRO Group 1

System/Evolution	K/A	SRO	KA Statement	Lev	Question Topic
295026	EA2.01	4.2	Suppression pool water temperature	S	Startup following high suppression pool temp required scram
	2.4.1	4.6	Knowledge of EOP entry conditions and immediate action steps.	S	EOP entry on high supp pool temp during testing
295030	EA2.01	4.2	Suppression pool level	S	Rapid depress during an ATWS
	EK3.03	3.7	RCIC operation: Plant-Specific	S	Consequences of running RCIC with low suppression pool level
295031	EK3.04	4.3	Steam cooling	S	Why water level allowed to go to -205 for steam cooling
295037	2.1.7	4.4	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	S	Commence injection during an ATWS with Rapid Depress
	EK3.03	4.5	Lowering reactor water level	S	Table 15 systems/LPCI injection during an ATWS
295038	EK3.02	4.2	System isolations	S	Isolation of systems during releases

Facility: Susquehanna   Exam Date: 5/10/99   Examination Level: SRO

Section Title Emergency and Abnormal Plant Evolutions

SRO Group 2

System/Evolution	K/A	SRO	KA Statement	Lev	Question Topic
295001	AA1.01	3.6	Recirculation system	S	Plant conditions when the actions required by ON-164-002 are applicable
	AK1.02	3.5	Power/flow distribution	S	Operating in Region II of Power/Flow Map
295002	AK2.05	2.7	Feedwater system	S	Reactor scram on low vacuum
295004	AA1.02	4.1	Systems necessary to assure safe plant shutdown	S	Loss of 125 VDC affect on paralleled diesel
295005	2.4.48	3.8	Ability to interpret control room indications to verify the status and operation of system, and understand how operator action s and directives affect plant and system conditions.	S	Overspeeding turbine actions
295012	AK2.02	3.7	Drywell cooling	S	Drywell cooling capabilities during a LOCA
295018	AK2.02	3.6	Plant operations	S	Limiting parameters on half isolation from loss of RPS Bus
295019	AA2.02	3.7	Status of safety-related instrument air system loads (see AK2.1 - AK2.19)	S	When scram required on loss of instrument air
295020	AK3.04	4.1	Reactor pressure response	S	Single MSIV closure at power effects
295021	AA1.04	3.7	Alternate heat removal methods	S	SRV operation on loss of SDC in Mode 3
295022	AK1.01	3.4	Reactor pressure vs. rod insertion capability	S	Loss of CRD actions during a startup
295028	EK1.01	3.7	Reactor water level measurement	S	Affects of increasing drywell temps on HPCI initiation
295029	EA2.03	3.5	Drywell/containment water level	S	Primary Containment water level measurements to determine if core covered (>TAF)
295032	2.2.3	3.3	(multi-unit) Knowledge of the design, procedural, and operational differences between units.	S	Unit differences on RCIC/HPCI operations in EOP-104/204
295033	EA1.03	3.8	Secondary containment ventilation	S	EO-104 actions to lower off-site release rates
295036	EA2.02	3.1	Water level in the affected area	S	Determination of Max Safe Water Levels in Secondary Containment
600000	EK3.04	3.4	Actions contained in the abnormal procedure for plant fire on site	S	Actions for a fire with Control Room Evacuation

Facility: Susquehanna **Date of Examination:** 05/10/99
Examination Level: SRO(I) **Operating Test No:**

System / JPM Title / Type Codes*	Safety Function	Planned Follow-up Questions: K/A/G - Importance - Description
1. RMCS/Take Actions For A Control Rod Double Notch/S,N	1	a. 201001K407 - 3.1/3.0 - PCV and FCV operational relationship
		b. 201001K110 - 2.8/2.8 - Control rod speed adjustments
2. HPCI/Place HPCI In CST To CST Mode - Steam Leak w/o Isolation/S,N,A	2	a. 206000K505 - 3.3/3.3 - Failed ramp generator
		b. 206000A217 - 3.9/4.3 - HPCI operation vs loss of feedwater heating
3. RHR-LPCI/Transfer From SDC To LPCI On Low Water Level/S,L,M	4	a. 295021K201 - 3.6/3.7 - Mode change/Tech Specs
		b. 205000G222 - 3.4/4.1 - Inop pressure switch - SDC Iso
4. RHR-SPC Mode/Place Suppression Pool Cooling In Service From RSDP/S,D	5	a. 219000A201 - 3.0/3.1 - RHR Pump vortex limits
		b. 295016K201 - 4.4/4.5 - RHR operations at RSDP
5. Diesel Gen/Synch "B" DG To 4.16KV Bus 2B-Runaway Diesel/S,M,A	6	a. 264000A210 - 3.9/4.2 - DG response to LOCA while paralleled
		b. 264000A404 - 3.7/3.7 - DG emergency stop actions and response
6. ADS/Respond To A Stuck Open Safety Relief Valve/S,M	3	a. 218000K106 - 3.9/3.9 - SRV operations with failed bellows
		b. 218000K601 - 3.9/4.1 - ECCS input to ADS logic
7. SGT/Perform Manual Startup Of SGTS & Vent The Drywell/S,M	9	a. 261000G112 - 2.9/4.0 - Inop SGTS vs Secondary Containment Integrity
		b. 261000A103 - 3.2/3.8 - Off-site doses at site boundary
8. SLC/Connect SLC Storage Tank To RCIC/M,P,R	1	a. 211000A205 - 3.1/3.4 - T.S. actions for SLC pump suction temperatures
		b. 211000A109 - 4.0/4.1 - Local pump operations
9. PCIS/Bypass MSIV And MSL Drain Isolation Signals/D,P	5	a. 223002K408 - 3.3/3.7 - One jumper not installed affect on isolation logic.
		b. 295037K306 - 3/8/4.1 - ES requirements vs plant conditions
10. RSCS/Bypass Control Rod In Rod Sequence Control System/P,M	7	a. 201004K301 - 3.3/3.4 - Rod movement with Inop RSCS
		b. 201004K501 - 3.6/3.0 - Fuel damage magnitude during rod drop accident

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

Facility: Susquehanna **Date of Examination:** 05/10/99
Examination Level: SRO(U) **Operating Test No:**

System / JPM Title / Type Codes*	Safety Function	Planned Follow-up Questions: K/A/G - Importance - Description
1. N/A		a.
		b.
2. N/A		a.
		b.
3. RHR-LPCI/Transfer From SDC To LPCI On Low Water Level/S,L,M	4	a. 295021K201 - 3.6/3.7 - Mode change/Tech Specs
		b. 205000G222 - 3.4/4.1 - Inop pressure switch - SDC Iso
4. RHR-SPC Mode/Place Suppression Pool Cooling In Service From RSDP/S,D	5	a. 219000A201 - 3.0/3.1 - RHR Pump vortex limits
		b. 295016K201 - 4.4/4.5 - RHR operations at RSDP
5. Diesel Gen/Synch "B" DG To 4.16KV Bus 2B-Runaway Diesel/S,M,A	6	a. 264000A210 - 3.9/4.2 - DG response to LOCA while paralleled
		b. 264000A404 - 3.7/3.7 - DG emergency stop actions and response
6. N/A		a.
		b.
7. N/A		a.
		b.
8. SLC/Connect SLC Storage Tank To RCIC/M,P,R	1	a. 211000A205 - 3.1/3.4 - T.S. actions for SLC pump suction temperatures
		b. 211000A109 - 4.0/4.1 - Local pump operations
9. N/A		a.
		b.
10. RSCS/Bypass Control Rod In Rod Sequence Control System/P,M	7	a. 201004K301 - 3.3/3.4 - Rod movement with Inop RSCS
		b. 201004K501 - 3.6/3.0 - Fuel damage magnitude during rod drop accident

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

ORIG SUBMITTAL

Susquehanna S.E.S.
1999 NRC Exam

Nuclear Department

In-Plant JPMs
Administrative JPMs

TAB I 1

PENNSYLVANIA POWER & LIGHT COMPANY
JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC 1-#8	0	05/10/99	217000	1
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Connect the SLC Storage Tank to the RCIC System

Completed By: **Reviews:**

C. J. Tyner	03/10/99		
Writer	Date	Instructor/Writer	Date

Approval:

Requesting Supv./C.A. Head	Date	Nuclear Training Supv.	Date

ALTERNATE PATH: NO TIME CRITICAL: NO RCA ENTRY: YES

TESTING METHOD: SIMULATE - PLANT

JPM SOURCE: Facility JPM 50.EO.005.102, Rev. 0 - modified to add the steps to actually inject, updated to current procedure revisions, added examiner Cues

Date of Performance:

	30 Min	
	Allowed Time (Min)	Time Taken (Min)

JPM Performed By

Last	First	M.I.	Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Ques #1: ( ) Satisfactory ( ) Unsatisfactory Ques #2: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

Signature	Typed or Printed

Comments:

REQUIRED TASK INFORMATION
JOB PERFORMANCE MEASURE
NRC 1-#8

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.
2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. ES-150-002, "Boron Injection Via RCIC", Rev. 11, Section 4.3

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. An ATWS condition exists

B. All efforts to insert the control rods have failed.

C. Both Recirc pumps have been tripped.

D. Suppression Pool temperature is 105° F

E. ADS has been inhibited.

F. SLC injection has failed.

G. EO-100-102, RPV Control, is being executed in conjunction with other required procedures.

V. INITIATING CUE

The Unit Supervisor directs you to line up the SLC storage tank to the RCIC System in accordance with ES-150-002.

PERFORMANCE CHECKLIST

Appl. To/JPM No.: NRC 1-#8

Candidate Name:___________________

Step	Action	Standard	Eval	Comments
	JPM Setup: • Obtain a copy of the latest revision of ES-150-002, and mark it up as if it was actually to be performed, and provide it to the Candidate with the Task Conditions/Initiating Cue Sheet.			
1.	Review Sections 1.0 through 3.0. Evaluator - tell Candidate that ES-150-001 has been evaluated and is not required	Reviews the purpose, required equipment, and the precautions and limitations sections of the procedure.		
2.	Ensure that Shift Supervision approval has been given to perform this procedure.	Notes that Section 4.1 is signed.		
3.	Obtain the required key. **Evaluator:** • The Candidate would need to obtain the SLC ES box key. • For purpose of this JPM, an ES key may be signed out from the Ops key locker with Shift Supervision approval. DO NOT remove a key from the ES box in the Shift Supervisor's office.	Obtains the following from Shift Supervision: • SLC ES box key		
4.	At RB Elevation 749', obtain the equipment to perform the connection. Evaluator - Inventory the equipment with the Candidate, then restore all equipment to the box and lock it. No equipment is to be removed.	Opens the RCIC ES box and obtains equipment.		

* - Critical Step # - Critical Sequence

Appl. To/JPM No.: NRC 1-#8 Candidate Name:________________

Step	Action	Standard	Eval	Comments
5.	Rotate pipe elbow on the downstream side of the SLC Flushing Drain 148F015. Evaluator - This elbow is located just prior to the floor drain near the pipe support between the pumps. CUE - When correct elbow located, cue that elbow connection has been loosened and the elbow is being rotated to the horizontal position facing South	Locates the pipe elbow. Loosens elbow connections and rotates the elbow to a horizontal position (Facing South).		
*6.	Install piping in the SLC Flushing Drain Line. CUE - When correct pipe selected, cue that connections are being made up, pipe is installed and tightened	Installs the two foot section of one inch pipe, taken from the RCIC ES box, into the elbow in the SLC Flushing Drain Line.		
*7.	Install the hose coupling on the pipe just installed. Cue - hose coupling installed on 2 foot pipe section, connections tightened	Installs a one inch double female pipe coupling, taken from the RCIC ES box, on the end of the two foot pipe that was just installed.		
8.	Place the noncollapsible hose in place. Cue - hose run to northeast stairwell and lowered down to RCIC room. Hose tied off as necessary.	Unreels the 300 feet of 1.5 inches noncollapsible hose down the northeast stairwell to the RCIC Room on RB Elevation 645'.		
*9.	Connect noncollapsible hose to SLC system. Cue - hose connected to SLC with hose clamps, all connections tightened.	Using both one foot hose clamps, taken from the RCIC ES box, fasten the noncollapsible hose to the pipe coupling installed in the two foot section of one inch pipe.		

* - Critical Step # - Critical Sequence

Appl. To/JPM No.: NRC 1-#8

Candidate Name:______________________

Step	Action	Standard	Eval	Comments
10.	Securely tie hose. **Cue** - hose securely tied,	Securely ties the hose, using the nylon rope obtained from the ES box. Hose not tied to snubbers.		
11.	Remove Cap From RCIC Supp Pool Suction Drain Valve 149012 **Cue** - when correct valve located, cap is removed from drain valve.	Removes the cap from the RCIC Supp Pool Suction Drn 149012.		
*12.	Install one-inch coupling to RCIC Supp Pool Suction Drain Valve 149012 **Cue** - coupling installed on 149012 and connection tightened	Installs a one-inch double female coupling, taken from the RCIC ES box, on RCIC Supp Pool Suction Drn Valve 149012.		
*13.	Connects 1.5 inch non-collapsible hose with hose clamp to pipe coupling at 149012 drain line **Cue** - hose connected to drain line, connection tightened	Hose connected to 149012 drain line		
14.	Informs Control Room that SLC is connected to RCIC **Cue** - Control Room acknowledges the call and informs you that RWCU is isolated, RCIC is injecting into the reactor and they are ready for Step 4.4	Calls Control Room and informs them that Sections 4.3.1 and 4.3.2 of ES-150-002 have been completed.		
*15.	Open SBLC Injection Pumps Suction Drain Valve 148F015 (El 749, Area 29) **Cue** - when correct valve located, valve repositioned fully counter-clockwise	Repositions F015 fully open		

* - Critical Step # - Critical Sequence

Appl. To/JPM No.: NRC 1-#8 Candidate Name:________________

Step	Action	Standard	Eval	Comments
16.	Check hose for leaks and flow restrictions **Cue** - no leaks or flow restrictions in the hose	Walks down hose and verifies no leaks, kinks, bends, etc.		
17.	Informs Control Room hose is pressurized **Cue** - Control Room acknowledges and has completed Step 4.4.3, they direct you to complete the lineup to inject	Calls Control Room and reports Step 4.4.2 completed		
18.	Verifies SBLC Storage Tank heaters in "Auto" (EL 749, Area 29) **Cue** - Heater switch is in "Auto"	Checks heater switch in "Auto"		
19.	Informs Control Room that flow is being initiated **Cue** - Control Room acknowledges	Calls Control Room and reports Step 4.4.5		
*20.	Opens RCIC Supp Pool Suction Drain (149012) (El 645, Area 28) **Cue** - When correct valve located, valve is positioned fully counter-clockwise	Repositions 149012 fully open		
*21.	Opens RCIC Pump Suct From Supp Pool Bypass (149019) (El645, Area 28) **Cue** - When correct valve located, valve is positioned fully counter-clockwise, Control Room reports lowering SBLC Storage Tank level	Repositions 149019 fully open		

* - Critical Step # - Critical Sequence

TASK CONDITIONS

A. An ATWS condition exists.

B. All efforts to insert the control rods have failed.

C. Both Recirc pumps have been tripped.

D. Suppression Pool temperature is 105° F

E. ADS has been inhibited.

F. SLC injection has failed.

G. EO-100-102, RPV Control, is being executed in conjunction with other required procedures.

INITIATING CUE

The Unit Supervisor directs you to line up the SLC storage tank to the RCIC System in accordance with ES-150-002.

TASK CONDITIONS

A. An ATWS condition exists.

B. All efforts to insert the control rods have failed.

C. Both Recirc pumps have been tripped.

D. Suppression Pool temperature is 105° F

E. ADS has been inhibited.

F. SLC injection has failed.

G. EO-100-102, RPV Control, is being executed in conjunction with other required procedures.

INITIATING CUE

The Unit Supervisor directs you to line up the SLC storage tank to the RCIC System in accordance with ES-150-002.

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#8 Candidate Name:________________________

QUESTION NO: __1__

Given the following Standby Liquid Control parameters taken from the logs:

- Storage tank level 4755 gallons
- Sodium Pentaborate concentration 13.8%
- Storage tank temperature 71 degrees F
- SLC Pump suction temperature 64 degrees F

What are the required actions for the given conditions?

EXPECTED ANSWER:

- SLC Pump suction temperature is out of specification low
- Declare both SLC subsystems Inoperable, restore at least one subsystem to Operable within 8 hours or be in Mode 3 within the next 12 hours
- Restore pump suction temperature to 69-69 degrees F (per Figure 3.1.7-2)
- Perform the biannual determination that the heat tracing piping between the storage tank and the pump suction is unblocked within 24 hours of restoring correct temperature

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 211000A205 3.1/3.4

REFERENCES: Unit 1 Tech Spec 3.1.7, Page 3.1-20

SY0-17 C-3, "Standby Liquid Control System", Rev. 2, LO - 4

Appl. To/JPM No: NRC 1-#8 Candidate Name:______________________

QUESTION NO: 2

The Standby Liquid Control quarterly flow verification is about to be run. How is the Reactor Water Cleanup System isolation and Squib Valves firing avoided during this test? Explain/verify your answer in prints.

EXPECTED ANSWER:

- The pumps are started and run from the local control station by holding the pushbutton depressed
- Pump starts with these pushbuttons bypass the RWCU isolation and the Squib Valve firing circuits.

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 211000A109 4.0/4.1

REFERENCES: SYD-17 C-3, "Standby Liquid Control System", Rev. 2, Figure 6, LO - 6.a & 14.d

TAS I 2

PENNSYLVANIA POWER & LIGHT COMPANY
JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC 1-#9	0	05/10/99	223002	5
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Bypass All MSIV Isolation Signals and MSL Drain Isolation Signals

Completed By: Reviews:

C. J. Tyner	03/10/99		
Writer	Date	Instructor/Writer	Date

Approval:

Requesting Supv./C.A. Head	Date	Nuclear Training Supv.	Date

ALTERNATE PATH: NO TIME CRITICAL: NO RCA ENTRY: NO

TESTING METHOD: SIMULATE - PLANT

JPM SOURCE: Facility JPM 84.EO.001.102, Rev. 0 - direct from source, updated to current procedure revisions, added examiner Cues

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Date of Performance:

	20 Min	
	Allowed Time (Min)	Time Taken (Min)

JPM Performed By

Last	First	M.I.	Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Ques #1: ( ) Satisfactory ( ) Unsatisfactory Ques #2: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

Signature	Typed or Printed

Comments:

REQUIRED TASK INFORMATION
JOB PERFORMANCE MEASURE
NRC 1-#9

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. ES-184-002, "Reopening MSIVs Bypassing Isolations", Rev. 6, Section 4.2.4

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. The Unit has had a failure to scram (ATWS)

B. The pressure control leg of EO-100-113, "Level/Power Control", has directed bypassing MSIV interlocks

C. The following actions have been completed from ES-184-002
- The handswitches for all MSIVs have been placed to CLOSE.
- Containment Instrument Gas has been restored in accordance with ES-184-002.
- Instrument Air has been restored in accordance with ON-118-001.
- RPS power has been restored in accordance with OP-158-001.
- The Circulating Water System has been started in accordance with OP-142-001.
- Section 4.2.3 is complete

D. 125 V DC is NOT available.

V. INITIATING CUE

The Unit Supervisor directs you to bypass ALL MSIV and MSL Drain isolation signals in accordance with ES-184-002.

Appl. To/JPM No.: NRC 1-#9

Candidate Name:_______________

Step	Action	Standard	Eval	Comments
	JPM Setup: Obtain a copy of ES-184-002 and mark it up as if Section 4.2.4 is actually to be performed and provide it to the Candidate.			
1.	Review Sections 1.0 through 3.0 and completed portions of 4.0	Reviews Sections 1.0, 2.0, 3.0 and completed portions of 4.0		
2.	Ensure Shift Supervision approval obtained to perform Section 4.2.4.	Observes Shift Supervision initials in Section 4.1giving approval to perform Section 4.2.4		
3.	Obtain the required jumpers. Evaluator - Have the Candidate show you the jumpers, but do not remove them from the SS box.	Obtains required jumpers from ES box in Shift Supervisor's office.		
*4.	Install jumper between terminal posts 11 and 13 on relay B21H-K7A. Evaluator - Panel 1C609, RPS Trip Sys A1/A2 NSS Shutoff Sys Panel, is located in the Upper Relay Room. Relay B21H-K7A is labeled CX/B21H-K7A and is located in 1C609 DIV 1 section inside the right door. It is in the second row of relays from the top on the far right. Refer to Attachment A of ES-184-002 for location of terminal posts 11 and 13. Cue - when correct panel and relay identified, jumper is installed between terminal posts 11 & 13	Correctly locates 1C609. Correctly identifies Relay B21H-K7A. Installs jumper between terminal posts 11 and 13.		

* - Critical Step # - Critical Sequence

Appl. To/JPM No.: NRC 1-#9 Candidate Name:______________________

Step	Action	Standard	Eval	Comments
*5.	Install jumper between terminal posts 11 and 13 on Relay B21H-K7C. Evaluator - Relay B21H-K7C is labeled AN/B21H-K7C and is located in 1C609 DIV 2 section inside the right door. It is in the second row of relays from the top on the far right. Refer to Attachment A of ES-184-002 for location of terminal posts 11 and 13. Cue - when correct panel and relay identified, jumper is installed between terminal posts 11 & 13	Correctly identifies Relay B21H-K7C. Installs jumper between terminal posts 11 and 13.		
*6.	Install jumper between terminal posts 11 and 13 on Relay B21H-K7B. Evaluator - Panel 1C611, RPS Trip Sys B1/B2 NSS Shutoff Sys Panel, is located in the Lower Relay Room. Relay B21H-K7B is labeled CX/B21H-K7B and is located in 1C611 DIV 1 section inside the right door. It is in the second row of relays from the top of the far right. Refer to Attachment A of ES-184-002 for location of terminal posts 11 and 13. Cue - when correct panel and relay identified, jumper is installed between terminal posts 11 & 13	Correctly locates 1C611. Correctly identifies Relay B21H-K7B. Installs jumper between terminal posts 11 and 13.		

* - Critical Step # - Critical Sequence

Appl. To/JPM No.: NRC 1-#9

Candidate Name:______________

Step	Action	Standard	Eval	Comments
*7.	Install jumper between terminal posts 11 and 13 on Relay B21H-K7D. **Evaluator** - Relay B21H-K7D is labeled AN/B21H-K7D and is located in 1C611 DIV 2 section inside the right door. It is in the second row of relays from the top on the far right. Refer to Attachment A of ES-184-002 for location of terminal posts 11 and 13. **Cue** - when correct panel and relay identified, jumper is installed between terminal posts 11 & 13	Correctly identifies Relay B21H-K7D. Installs jumper between terminal posts 11 and 13.		
8.	Inform Control Room all MSIV and MSL Drain Valve Isolation Signals have been bypassed **Evaluator** - acknowledge report as Control Room	Calls Control Room and reports Section 4.2.4 completed.		

* - Critical Step # - Critical Sequence

TASK CONDITIONS:

A. The Unit has had a failure to scram (ATWS)

B. The pressure control leg of EO-100-113, "Level/Power Control", has directed bypassing MSIV interlocks

C. The following actions have been completed
- The handswitches for all MSIVs have been placed to CLOSE.
- Containment Instrument Gas has been restored in accordance with ES-184-001.
- Instrument Air has been restored in accordance with ON-118-001.
- RPS power has been restored in accordance with OP-158-001.
- The Circulating Water System has been started in accordance with OP-142-001.
- Section 4.2.3 is complete

D. 125 V DC is NOT available.

INITIATING CUE:

The Unit Supervisor directs you to bypass ALL MSIV and MSL Drain isolation signals in accordance with ES-184-002.

TASK CONDITIONS:

A. The Unit has had a failure to scram (ATWS)

B. The pressure control leg of EO-100-113, "Level/Power Control", has directed bypassing MSIV interlocks

C. The following actions have been completed
   - The handswitches for all MSIVs have been placed to CLOSE.
   - Containment Instrument Gas has been restored in accordance with ES-184-001.
   - Instrument Air has been restored in accordance with ON-118-001.
   - RPS power has been restored in accordance with OP-158-001.
   - The Circulating Water System has been started in accordance with OP-142-001.
   - Section 4.2.3 is complete

D. 125 V DC is NOT available.

INITIATING CUE:

The Unit Supervisor directs you to bypass ALL MSIV and MSL Drain isolation signals in accordance with ES-184-002.

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#9 Candidate Name:____________________

QUESTION NO: 1

If the jumper between terminal posts 11 & 13 in the back of 1C609 had NOT been installed (Step 4.2.4.1.(2)) and reactor water level subsequently lowered to -129 inches, what would be the response of the MSIVs? Confirm your answer utilizing logic prints. Assume the other 3 jumpers were correctly installed.

EXPECTED ANSWER:

The MSIVs should remain open. Installing three jumpers should still defeat the needed "one-out-of-two-taken-twice" logic for an isolation.

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 223002K408 3.3/3.7

REFERENCES:

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#9 Candidate Name:______________________________

QUESTION NO: _2_

Under what specific plant conditions is opening the Main Steam Isolation Valves authorized with a fuel failure or steam line break present?

EXPECTED ANSWER:

- Conditions require a Rapid Depressurization and less than 4 SRV can be opened
- Conditions require RPV venting

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 295037K306 3.8/4.1

REFERENCES: ES-184-002, "Reopening MSIVs Bypassing Isolations", Rev. 6, Section 6.3.8, Page 29

JPS I 3

PENNSYLVANIA POWER & LIGHT COMPANY
JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC 1-#10	0	05/10/99	201004	7
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Bypass of Control Rod - Rod Sequence Control System (RSCS)

Completed By:

C. J. Tyner
Writer

03/10/99
Date

Reviews:

Instructor/Writer ____________ Date ____________

Approval:

Requesting Supv./C.A. Head ____________ Date ____________ Nuclear Training Supv. ____________ Date ____________

ALTERNATE PATH: NO TIME CRITICAL: NO RCA ENTRY: NO

TESTING METHOD: SIMULATE - PLANT

JPM SOURCE: Facility JPM 56.OP.008.101, Rev. 0 - modified to different control rod, updated to latest procedure revisions, added examiner Cues

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Date of Performance: ____________

10 Min
Allowed Time (Min)

Time Taken (Min) ____________

JPM Performed By

Last ____________ First ____________ M.I. ____________ Employee #/S.S. # ____________

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Ques #1: ( ) Satisfactory ( ) Unsatisfactory Ques #2: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

Signature ____________ Typed or Printed ____________

Comments:

REQUIRED TASK INFORMATION
JOB PERFORMANCE MEASURE
NRC 1-#10

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. OP-156-002, "Rod Sequence Control System (RSCS)", Rev. 8, Section 3.2

B. NDAP-QA-0338, "Reactivity Management and Controls Program", Rev. 5, Attachment I

III. REACTIVITY MANIPULATIONS

None

IV. TASK CONDITIONS

A. A startup is in progress on Unit 1.

B. Control Rod 34-47 has experienced a RPIS failure.

C. There are no control rods bypassed on the RSCS Cabinet 1C649.

D. Form NDAP-QA-0338, Attachment I, has been completed, authorizing Control Rod 34-47 to be bypassed in RSCS.

V. INITIATING CUE

The Unit Supervisor directs you to bypass Control Rod 34-47 in the Rod Sequence Control System (RSCS)

PERFORMANCE CHECKLIST

Appl. To/JPM No.: NRC 1-#10 Candidate Name:_______________

Step	Action	Standard	Eval	Comments
	Simulator Setup: • Markup copy of NDAP-QA-0338, Attachment I, to authorize bypassing control Rod 34-47			
1.	Obtain a controlled copy of OP-156-002.	Controlled copy obtained.		
2.	Select the correct section to perform.	Selects Section 3.2		
3.	Review the prerequisites. **Evaluator** Inform Candidate all prerequisites have been met.	Ensures all prerequisites have been met.		
4.	Review the precautions.	Precautions reviewed		
*5.	Determine Binary Coordinate Code for control rod 34-47 by using the Fault Map on Analyzer Section of Rod Drive Control Cabinet 1C616 or Attachment B of this procedure.	Determine Binary Coordinate Code for control rod 34-47 is 01010 01101		
6.	Open RSCS Bypass Switch Card Cover in RSCS Cabinet 1C649	RSCS Bypass Switch Card Cover opened		
*7.	Select first Bypass Switch Card not in use and position Bypass Ident Select Switches in proper Binary Coordinate Code positions. **Cue** - When card and toggle switches located, cue Candidate that each is placed in the position stated	Code is 01010 01101 with switches positioned left for "0" and right for "1"		

* - Critical Step # - Critical Sequence

Appl. To/JPM No.: NRC 1-#10 Candidate Name:______________

Step	Action	Standard	Eval	Comments
*8.	Place bypass switch on Bypass Switch Card in BYPASS position. **Cue** - when switch located, cue Candidate switch is positioned as stated, red light is "on"	Positions Bypass Switch to "Bypass" (to the right), observes red light on		
9.	Direct Unit PCO to perform the following: • Using RED Display Control, check RED LED at core location 34-47 illuminated. • Select Control Rod 34-47 at Control Rod Select pushbuttons, and verify withdraw and insert blocks are clear. **Evaluator** - Inform Candidate that Control Room RSCS Display red LED is illuminated, that control rod 34-47 has been selected and the insert and withdraw blocks are clear.	Directs Unit PCO to perform Steps e, f & g of Section 3.2		

* - Critical Step # - Critical Sequence

TASK CONDITIONS

A. A startup is in progress on Unit 1.

B. Control Rod 34-47 has experienced a RPIS failure.

C. There are no control rods bypassed on the RSCS Cabinet 1C649.

D. Form NDAP-QA-0338, Attachment I, has been completed, authorizing Control Rod 34-47 to be bypassed in RSCS.

INITIATING CUE

The Unit Supervisor directs you to bypass Control Rod 34-47 in the Rod Sequence Control System (RSCS).

TASK CONDITIONS

A. A startup is in progress on Unit 1.

B. Control Rod 34-47 has experienced a RPIS failure.

C. There are no control rods bypassed on the RSCS Cabinet 1C649.

D. Form NDAP-QA-0338, Attachment I, has been completed, authorizing Control Rod 34-47 to be bypassed in RSCS.

INITIATING CUE

The Unit Supervisor directs you to bypass Control Rod 34-47 in the Rod Sequence Control System (RSCS).

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#10 Candidate Name:______________________

QUESTION NO: 1

Unit 1 reactor power is 15% with a startup in progress. The Rod Sequence Control System has been declared Inoperable.

What are the choices available to the operator regarding continued control rod movement?

EXPECTED ANSWER:

- Suspend control rod movement except by scram
- OR bypass RSCS and verify all rod movements by a second licensed operator or other qualified member of the technical staff
- OR bypass RSCS and verify RWM Operable IAW LCO 3.3.2.1

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 201004K301 3.3/3.4

REFERENCES: Unit TRM 3.1.5, Page 3.1-12

SY017 K-4, "Rod Sequence Control System", Rev. 1, LO - 7 & 8

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#10 Candidate Name:____________________

QUESTION NO: 2

The Rod Sequence Control System together with the Rod Worth Minimizer are designed to limit control rod worth such that the "worst possible" rod drop accident would result in no more than 280 calories of energy deposited per gram of fuel.

What will be the effect of this level of energy disposition on the fuel?

EXPECTED ANSWER:

280 cal/gm will melt the Uranium Dioxide fuel but will only perforate the cladding, not melt it. Thus fuel damage can occur for this rod drop accident but fission product release is minimized.

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 201004K501 3.6/4.0

REFERENCES: SY017 K-4, "Rod Sequence Control System", Rev. 1, Section II.A, Pages 1 & 2

TR(?) A1

PENNSYLVANIA POWER & LIGHT COMPANY
ADMIN JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC Admin A.1 #1	0	05/10/99	N/A	N/A
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Calculate A Cooldown Rate In Accordance With SO-100-011

Completed By:

Reviews:

C. J. Tyner	03/20/99		
Writer	Date	Instructor/Writer	Date

Approval:

Requesting Supv./C.A. Head	Date	Nuclear Training Supv.	Date

ALTERNATE PATH: N/A TIME CRITICAL: NO LOW POWER/SHUTDOWN: N/A

TESTING METHOD: PERFORM - SIMULATOR/PLANT

JPM SOURCE: NEW

Date of Performance:

	10 Min	
	Allowed Time (Min)	Time Taken (Min)

JPM Performed By:

Last	First	M.I.	Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

Signature	Typed or Printed

Comments:

REQUIRED TASK INFORMATION
ADMIN JOB PERFORMANCE MEASURE
NRC Admin A.1 #1

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. SO-100-011, "Reactor Vessel Temperature And Pressure Recording", Rev. 12, Section 6.1

B. Evaluator - See attached copy of SO-100-011, Attachment A filled out as expected.

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. Unit 1 is shutdown and is performing a normal cooldown and depressurization

B. The plant process computer is not available for taking cooldown data

V. INITIATING CUE

The Unit Supervisor directs you to take plant cooldown data and to determine if the Technical Specification cooldown rate has been violated.

JPM Setup - Ensure a copy of SO-100-011 and blank SO-100-11 Attachment A are available for Candidate.

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC Admin A.1 #1

Candidate Name:______________

Step	Action	Standard	Eval	Comments
1.	Obtain a controlled copy of SO-100-011	Controlled copy obtained		
2.	Selects correct section to perform **Evaluator** - provide copy of Attachment A when Candidate locates controlled copy	Selects Section 6.1 and Attachment A		
3.	Records current date and time	Records date/time on Attachment A		
4.	Records initial set of plant parameters at time "0" from panel indications **Cue** - When correct recorder and panel identified for each data point provide the following to the Candidate at time "0" • Recirc Loop "A" - 435 deg F • Recirc Loop "B" - 435 deg F • Bottom Head Drain - 402 deg F • Reactor pressure - 485 psig	Records the following parameters: • Recirc Loop "A" temperature (TR-B31-1R650 on C652) • Recirc Loop "B" temperature (TR-B31-1R650 on C652) • Reactor Vessel Bottom Head Drain temperature (TR-B21-1R006 on C007) • Reactor pressure (PI-C32-1R605 on C652)		
*5.	Calculate and record steam dome temperature **Evaluator** - 485 psig + 15 psi = 500 psia = 467 deg F **Cue** - When steam dome temp calculated and recorded cue candidate that 15 minutes have passed	Notes direct steam dome temperature not available, calculates temperature using steam tables and reactor pressure, records approx 467 deg F		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC Admin A.1 #1

Candidate Name:_______________

Step	Action	Standard	Eval	Comments
6.	Records plant parameters at time "15" from panel indications **Cue** - When correct recorder and panel identified for each data point provide the following to the Candidate at time "15" • Recirc Loop "A" - 425 deg F • Recirc Loop "B" - 425 deg F • Bottom Head Drain - 389 deg F • Reactor pressure - 395 psig	Records the following parameters: • Recirc Loop "A" temperature (TR-B31-1R650 on C652) • Recirc Loop "B" temperature (TR-B31-1R650 on C652) • Reactor Vessel Bottom Head Drain temperature (TR-B21-1R006 on C007 • Reactor pressure (PI-C32-1R605 on C652)		
*7.	Calculate and record steam dome temperature **Evaluator** - 395 psig + 15 psi = 410 psia = 447 deg F	Notes direct steam dome temperature not available, calculates temperature using steam tables and reactor pressure, records approx 447 deg F		
*8.	Calculate and record temperature changes for first 15 minutes of cooldown **Cue** - When delta T's calculated and recorded cue candidate that 15 additional minutes have passed	Calculates/records delta T's from time "0" to time "15" as follows: • Recirc Loop "A" - 10 deg F • Recirc Loop "B" - 10 deg F • Bottom head drain - 13 deg F • Steam dome - 20 deg F		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC Admin A.1 #1

Candidate Name:____________________

Step	Action	Standard	Eval	Comments
9.	Records plant parameters at time "30" from panel indications Cue - When correct recorder and panel identified for each data point provide the following to the Candidate at time "30" • Recirc Loop "A" - 417 deg F • Recirc Loop "B" - 417 deg F • Bottom Head Drain - 378 deg F • Reactor pressure - 320 psig	Records the following parameters: • Recirc Loop "A" temperature (TR-B31-1R650 on C652) • Recirc Loop "B" temperature (TR-B31-1R650 on C652) • Reactor Vessel Bottom Head Drain temperature (TR-B21-1R006 on C007 • Reactor pressure (PI-C32-1R605 on C652)		
*10.	Calculate and record steam dome temperature Evaluator - 320 psig + 15 psi = 335 psia = 428 deg F	Notes direct steam dome temperature not available, calculates temperature using steam tables and reactor pressure, records approx 428 deg F		
*11.	Calculate and record temperature changes for second 15 minutes of cooldown	Calculates/records delta T's from time "15" to time "30" as follows: • Recirc Loop "A" - 8 deg F • Recirc Loop "B" - 8 deg F • Bottom head drain - 11 deg F • Steam dome - 19 deg F		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC Admin A.1 #1 Candidate Name:____________

Step	Action	Standard	Eval	Comments
*12.	Confirm compliance with TS 3.4.10.1 **Evaluator** - SO-100-001 Step 6.1.4 at bottom of page 6 directs verifying TS compliance every half hour. Candidate verifying TS compliance after just first 15 minutes of data is NOT satisfactory. In addition, Not (1) at top of page 7 states that Steam Dome Temperature should be used to "best determine" cooldown rate. Candidate should use the change in calculated steam dome temperature from time "0" to time "30" to determine if TS limit of >100 deg in any one hour has been violated.	Calculates steam dome temperature change from time "0" to time "30". Determines change is 39 deg. Determines current cooldown rate is 78 deg/hour. Reports to US that cooldown is 78 deg/hour and is less than TS limit .		

* - Critical Step # - Critical Sequence

TASK CONDITIONS:

A. Unit 1 is shutdown and is performing a normal cooldown and depressurization

INITIATING CUE:

The Unit Supervisor directs you to take plant cooldown data and to determine if the Technical Specification cooldown rate has been violated.

TASK CONDITIONS:

A. Unit 1 is shutdown and is performing a normal cooldown and depressurization

INITIATING CUE:

The Unit Supervisor directs you to take plant cooldown data and to determine if the Technical Specification cooldown rate has been violated.

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PENNSYLVANIA POWER & LIGHT COMPANY
ADMIN JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC Admin A.1 #2	0	05/10/99	N/A	N/A
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Authorize Bypassing Rod Worth Minimizer In Accordance With NDAP-QA-0338

Completed By: Reviews:

C. J. Tyner — Writer; 03/20/99 — Date; ____ Instructor/Writer; ____ Date

Approval:

____ Requesting Supv./C.A. Head; ____ Date; ____ Nuclear Training Supv.; ____ Date

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ALTERNATE PATH: N/A TIME CRITICAL: NO LOW POWER/SHUTDOWN: N/A

TESTING METHOD: PERFORM - SIMULATOR/PLANT

JPM SOURCE: NEW

Date of Performance: ____

10 Min — Allowed Time (Min) ____ Time Taken (Min)

JPM Performed By:

____ Last First M.I. ____ Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

____ Signature ____ Typed or Printed

Comments:

REQUIRED TASK INFORMATION
ADMIN JOB PERFORMANCE MEASURE
NRC Admin A.1 #2

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. NDAP-QA-0338, "Reactivity Management And Controls Program", Rev. 5, Section 6.8

B. NDAP-QA-0338, Attachment I

C. NDAP-QA-0312, Attachment C

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. Unit 1 is at 8% with power ascension to 100% in progress

B. Control rod 50-35 has just been withdrawn from Notch "08" to Notch "12" per step A2-306

C. When rod 50-27 was withdrawn from Notch "08" to "12" per Step A2-307, a Rod Block alarm was received and rod motion stopped

D. RSCS indications show it is not generating the rod block and Rod Position Indication is Operable

E. The Rod Worth Minimizer Withdraw Block status light is illuminated

F. All rods have been verified to be in the pull sheet required positions

G. The Reactor Engineer has verified the pull sheet to be correct

H. Troubleshooting has determined that a RWM failure has caused the rod block and that there have been no computer problems

V. INITIATING CUE

The Shift Supervisor directs you to determine if RWM bypassing is allowed for these conditions.

<u>Evaluator</u> - once Candidate has determined the RWM can be bypassed, direct completion the required documentation and any additional actions.

<u>JPM Setup</u> - Ensure a copy of NDAP-QA-0338, blank NDAP-QA-0338 Attachment I and LCO Report form are available for Candidate.

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC Admin A.1 #2

Candidate Name:______________________

Step	Action	Standard	Eval	Comments
1.	Obtain controlled copy of NDAP-QA-0338	Controlled copy obtained		
2.	Selects correct section to perform **Evaluator** - provide copy of Attachment I when Candidate locates controlled copy	Selects Section 6.8 and Attachment I		
*3.	Determines Rod Worth Minimizer can be bypassed **Evaluator** - see attached copy of Attachment I **Evaluator** - once Candidate has determined RWM can be bypassed, cue completion of documentation and any additional required actions.	Completes NDAP-QA-0338 Section 6.8 and Attachment I, determines RWM can be bypassed, informs Shift Supervisor		
*4.	Authorize bypassing RWM	Checks authorization to bypass RWM, signs/dates Attachment I		
*5.	Refers to Tech Spec 3.3.2.1, Action C	Determines startup can continue if all rod movements verified to be in accordance with BPWS by 2nd licensed operator, STA or RE AND if the RWM has not been Inop for a startup in the last calendar year.		
*6.	Completes LCO Report Form for LCO 3.3.2.1 **Evaluator** - see attached copy of LCO Report Form	LCO Report Form completed		

* - Critical Step # - Critical Sequence

TASK CONDITIONS:

A. Unit 1 is at 8% with power ascension to 100% in progress

B. Control rod 50-35 has just been withdrawn from Notch "08" to Notch "12" per step A2-306

C. When rod 50-27 was withdrawn from Notch "O8" to "12" per Step A2-307, a Rod Block alarm was received and rod motion stopped

D. RSCS indications show it is not generating the rod block and Rod Position Indication is Operable

E. The Rod Worth Minimizer Withdraw Block status light is illuminated

F. All rods have been verified to be in the pull sheet required positions

G. The Reactor Engineer has verified the pull sheet to be correct

H. Troubleshooting has determined that a RWM failure has caused the rod block and that there have been no computer problems

INITIATING CUE:

The Shift Supervisor directs you to determine if RWM bypassing is allowed for these conditions.

TASK CONDITIONS:

A. Unit 1 is at 8% with power ascension to 100% in progress

B. Control rod 50-35 has just been withdrawn from Notch "08" to Notch "12" per step A2-306

C. When rod 50-27 was withdrawn from Notch "O8" to "12" per Step A2-307, a Rod Block alarm was received and rod motion stopped

D. RSCS indications show it is not generating the rod block and Rod Position Indication is Operable

E. The Rod Worth Minimizer Withdraw Block status light is illuminated

F. All rods have been verified to be in the pull sheet required positions

G. The Reactor Engineer has verified the pull sheet to be correct

H. Troubleshooting has determined that a RWM failure has caused the rod block and that there have been no computer problems

INITIATING CUE:

The Shift Supervisor directs you to determine if RWM bypassing is allowed for these conditions.

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PENNSYLVANIA POWER & LIGHT COMPANY
ADMIN JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC Admin A.2	0	05/10/99	N/A	N/A
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Determine Plant Equipment Operability And Implement Tech Spec 3.0.3 Including Documentation/Reports

Completed By: **Reviews:**

C. J. Tyner ____ 03/20/99 ____ ________ ________
Writer Date Instructor/Writer Date

Approval:

________ ________ ________ ________
Requesting Supv./C.A. Head Date Nuclear Training Supv. Date

ALTERNATE PATH: N/A TIME CRITICAL: NO LOW POWER/SHUTDOWN: N/A

TESTING METHOD: PERFORM - SIMULATOR/PLANT

JPM SOURCE:

Date of Performance:

________ 15 Min ________
Allowed Time (Min) Time Taken (Min)

JPM Performed By:

________________ ________
Last First M.I. Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

________ ________
Signature Typed or Printed

Comments:

REQUIRED TASK INFORMATION
ADMIN JOB PERFORMANCE MEASURE
NRC Admin A.2

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. SO-159-002, "Monthly Operability Check Of Suppression Chamber Drywell Vacuum Relief Breaker Valves", Rev. 8

B. Unit Tech Spec 3.6.1.6 and 3.0.3

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. During a Recirc runback to the #1 Limiter, the "D" Safety Relief Valve momentarily opened and immediately reclosed

B. The appropriate operator actions for the runback and SRV opening were taken.

C. SO-159-002, "Monthly Operability Check Of Suppression Chamber Drywell Vacuum Relief Breaker Valves", was performed within 2 hours of the SRV opening

V. INITIATING CUE

You are directed to review the provided Checksheet 1 for Attachment A of SO-159-002, complete the Data Sheet for Attachment A and take the appropriate actions including any required documentation.

JPM Setup - Ensure a copy of SO-159-002 and SO-159-002 Checksheet 1 Attachment A are filled out per attached and a blank Data Sheet are available for Candidate.

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC Admin A.2

Candidate Name:____________________

Step	Action	Standard	Eval	Comments
1.	Reviews provided SO-159-002 Checksheet 1 Evaluator - if Candidate asks, the operator was directed to complete the entire surveillance even after the first inop vacuum breaker was discovered	Reviews Checksheet 1 Data		
*2.	Determines Vacuum Breaker Relief Valves Operability	Determines both Vacuum Breaker Relief Valves in Downcomers "B" and "E" did not open during test, declares both pairs of valves Inoperable		
*3.	Completes SO-159-002 Attachment A Data Sheet	Circles "No" for Acceptance Criteria 1 and 2 and initials in "Confirm" block, circles "Yes" for Required Action 1 and "No" for 2 & 3 and initials in "Confirm" block, may note in remarks that two pairs of vacuum breakers did not open		
*4.	Refers to TS 3.6.1.6	Declares Inboard and Outboard Vacuum Breaker Relief Valves for both the "B" and "E" Downcomer Inoperable, notes no applicable action statements for these conditions, determines TS 3.0.3 applicable		

* - Critical Step   # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC Admin A.2

Candidate Name:______________________

Step	Action	Standard	Eval	Comments
*5.	Refers to TS 3.0.3 and initiates actions to comply with 3.0.3 **Evaluator** - must start TS 3.0.3 actions within one hour	Required to initiate actions within 1 hour to be in Mode 2 within 7 hours, Mode 3 within 13 hours and Mode 4 within 37 hours		
6.	Directs actions for plant shutdown	• Direct Unit PCO to obtain and review GO-100-004, "Plant Shutdown to Minimum Power" • Contract RE for control rod shutdown sequence		
7.	Contact maintenance	• Initiate WA for Vacuum Breaker troubleshoot/repair • Inform maintenance that Unit is in Shutdown LCO		
8.	Make notifications	Notify: • Duty Manager • NRC • PCC • Operations Manager		
9.	Complete LCO Log Sheet	Completes LCO Log Sheet IAW NDAP-QA-0302		

* - Critical Step # - Critical Sequence

TASK CONDITIONS:

A. During a Recirc runback to the #1 Limiter, the "D" Safety Relief Valve momentarily opened and immediately reclosed

B. The appropriate operator actions for the runback and SRV opening were taken.

C. SO-159-002, "Monthly Operability Check Of Suppression Chamber Drywell Vacuum Relief Breaker Valves", was performed within 2 hours of the SRV opening

INITIATING CUE:

You are directed to review the provided Checksheet 1 for Attachment A of SO-159-002, complete the Data Sheet for Attachment A and take the appropriate actions including any required documentation.

TASK CONDITIONS:

A. During a Recirc runback to the #1 Limiter, the "D" Safety Relief Valve momentarily opened and immediately reclosed

B. The appropriate operator actions for the runback and SRV opening were taken.

C. SO-159-002, "Monthly Operability Check Of Suppression Chamber Drywell Vacuum Relief Breaker Valves", was performed within 2 hours of the SRV opening

INITIATING CUE:

You are directed to review the provided Checksheet 1 for Attachment A of SO-159-002, complete the Data Sheet for Attachment A and take the appropriate actions including any required documentation.

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SUSQUEHANNA NRC EXAM ADMIN QUESTIONS

CANDIDATE: ____________________ **DOCKET:** ______________ **DATE:** ______________

QUESTION: A.3 #1

The Liquid Radwaste Radiation Monitor has been determined to be Inoperable. What are the limitations and restrictions for liquid releases for these conditions?

ANSWER:

The Laundry Drain Sample Tank CANNOT be released. Other tanks (Liquid Radwaste Sample and Distillate Sample tanks) can be released for up to 14 days provided:

- A minimum of two independent tank samples are drawn and analyzed IAW TRO 3.11.1.1
- The release rate calculations are independently verified
- The release check-off list (discharge valve lineup) is performed and independently verified

RESPONSE:

SAT ______ **UNSAT** ______ **K/A NUMBER:** 268000G303 1.8/2.9

REFERENCES: OP-069-050, "Release Of Liquid Radioactive Waste", Rev. 21, Sections, 3.1.3.h, 3.2.3.h and 3.3.3.h, Pages 7, 33 and 61

NDAP-QA-0310, "Liquid Effluent Release", Rev. 3, Section 6.2, Page 9

Unit 1 TRM 3.11.1.4, Action B, Page 3.11-9

QUESTION: A.3 #2

Unit 1 is ready to perform a release of the "A" and "B" LRW Sample Tanks. Total Site Blowdown instrumentation is inoperable. The Unit 2 cooling tower basin is drained.

How is the minimum blowdown flow of 5500 gpm assured during the release of this tank for these conditions?

ANSWER:

Must ensure that Unit 1 cooling tower blowdown is at least 5500 gpm. Done by closing the Cooling Tower Blowdown Valve then reopening it to a position specified. Once at this position, the desired flow (from graph) is compared with indicated flow. If the indicated flow is above the graph, the instrumentation is providing valid indication..

RESPONSE:

SAT ______ **UNSAT** ______ **K/A NUMBER:** 268000G306 2.1/3.1

REFERENCES: OP-069-050, "Release Of Liquid Radioactive Waste", Rev. 21, Sections, 3.3.9, 3.3.10 and **Attachment F, Pages 68-70 and 168-174**

JOG A.5

PENNSYLVANIA POWER & LIGHT COMPANY
ADMIN JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC Admin A.4	0	05/10/99	N/A	N/A
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Classify And Make Protective Action Recommendations For A General Emergency

Completed By: **Reviews:**

C. J. Tyner ______ 03/20/99 ______ ______
Writer Date Instructor/Writer Date

Approval:

______ ______ ______ ______
Requesting Supv./C.A. Head Date Nuclear Training Supv. Date

ALTERNATE PATH: N/A TIME CRITICAL: YES LOW POWER/SHUTDOWN: N/A

TESTING METHOD: PERFORM - SIMULATOR/PLANT

JPM SOURCE: Facility JPM 9.100.01.081, Rev. 00 - Modified to provide new plant parameters for GE classification, updated to latest procedure revisions

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Date of Performance:

______ ______ ______
Allowed Time (Min) Time Taken (Min)

JPM Performed By:

______ ______
Last First M.I. Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

______ ______
Signature Typed or Printed

Comments:

REQUIRED TASK INFORMATION
ADMIN JOB PERFORMANCE MEASURE
NRC Admin A.4

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. EP-PS-100, "Emergency Director, Control Room", Rev. 13

B. EP-PS-126, "Control Room Communicator", Rev. 14

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. Unit 1 has experienced a MSIV closure from 100% power

B. Control rods did NOT insert and all methods of inserting rods have been unsuccessful

C. The Scram Discharge Volume did not isolate on the scram signal

D. Both Standby Liquid Control Squib Valves failed to fire

E. Reactor power is 43%

F. Suppression pool water temperature is 195 degrees F

G. Containment Rad Monitors are reading 1 rem/hour

V. INITIATING CUE

You are directed to classify this event and take appropriate actions IAW the Emergency Plan.

<u>JPM Setup</u> - Ensure a copy of EP-PS-100 and EP-PS-126, blank ENR and PAR forms and blank Notification Matrix are available for Candidate.

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC Admin A.4

Candidate Name:____________________

Step	Action	Standard	Eval	Comments
	Evaluator - After the Candidate has the task conditions, the initiating cue and understands the task, state that steps within this JPM are TIME CRITICAL. **Evaluator** - Note start time for event classification.			
*1.	Classify the emergency within 15 minutes for the given conditions. **Evaluator** - GE classification must be made within 15 minutes of start time	• Refers to EP-PS-0100 Emergency Director Tab 6 • Evaluate Unit conditions • Declares General Emergency IAW EAL #11.4 • Classification made within 15 minutes of JPM start		
2.	Document and communicate the classification	Refers to Tab "E" for the following: • Announce to CR that you are ED, a GE has been declared and the time and date of the classification. • Appoint a CR Communicator and direct performance of EP-PS-126 • Direct CR Communicator to make page announcement of classification • Initiate Accountability and Site Evacuation of non-essential personnel		
3.	**Cue** - Acknowledge CR announcement as US **Cue** - Acknowledge direction as Communicator			

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC Admin A.4

Candidate Name:_______________

Step	Action	Standard	Eval	Comments
*4.	Make public protective action recommendation within 15 minutes of General Emergency declaration. **Evaluator** - PAR shall be made within 15 minutes of GE classification time, see attached PAR form **Cue** - acknowledge as communicator	Refers to Tab 7 for the following: • Completes Tab 7 flow chart and determines that PAR should be Shelter in a 0 -10 mile radius • Completes PAR form per Tab 11 • Directs CR Communicator to notify PEMA EOC of PAR		
5.	Activate the Emergency Response Organizations	Directs CR Communicator to notify SCC to activate NERO: • Notifies HP and Chemistry • EOF Staff • OSC Staff • Duty Manager • Recovery Manager		
*6.	Complete Emergency Notification Report within 15 minutes. **Cue** - You are directed to complete the ENR **Evaluator** - See attached ENR, this step is critical for filling out the form, not meeting the 15 minute requirements since the Candidate is doing all the work alone **Cue** - When ENR completed, cue Candidate to demonstrate how to make the notifications.	Directs CR Communicator to complete ENR. • Completes ENR per Tab 11		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC Admin A.4

Candidate Name:________________________

Step	Action	Standard	Eval	Comments
7.	Make Emergency Notifications **Evaluator** - once notifications made, cue Candidate the JPM is complete.	Refers to EP-PS-126, Tab 4 and/or the flowchart and makes all required notifications.		

* - Critical Step # - Critical Sequence

TASK CONDITIONS:

A. Unit 1 has experienced a MSIV closure from 100% power

B. Control rods did NOT insert and all methods of inserting rods have been unsuccessful

C. The Scram Discharge Volume did not isolate on the scram signal

D. Both Standby Liquid Control Squib Valves failed to fire

E. Reactor power is 43%

F. Suppression pool water temperature is 195 degrees F

G. Containment Rad Monitors are reading 1 rem/hour

INITIATING CUE:

You are directed to classify this event and take appropriate actions IAW the Emergency Plan.

TASK CONDITIONS:

A. Unit 1 has experienced a MSIV closure from 100% power

B. Control rods did NOT insert and all methods of inserting rods have been unsuccessful

C. The Scram Discharge Volume did not isolate on the scram signal

D. Both Standby Liquid Control Squib Valves failed to fire

E. Reactor power is 43%

F. Suppression pool water temperature is 195 degrees F

G. Containment Rad Monitors are reading 1 rem/hour

INITIATING CUE:

You are directed to classify this event and take appropriate actions IAW the Emergency Plan.

ORIG. SUBMITTAL

Susquehanna S.E.S.
1999 NRC Exam

Nuclear Department

JPMs

Final Summary
5/3/99

SUSQUEHANNA NRC JPM EXAM
POST SUBMITTAL CHANGES

NOTE: NORMAL type indicates exam changes made from additional facility validation completed AFTER the initial exam submittal to the NRC.
BOLD type indicates exam changes made based upon the NRC comments per telcon on 04/15/99.
ITALICIZED type indicated exam changes made after Rev. 1 submittal to the NRC on 04/22/99.
<u>UNDERLINED type indicates exam changes made based upon NRC comments made during their prep week 04/26/99.</u>

SRO-I Outline - Changed JPM 1-#3 from "Transferring From SDC To LPCI On Low Water Level" to "Transferring Operating RHR Pumps While In Shutdown Cooling". Also changed system tested from RHR-LPCI Mode to Shutdown Cooling.

Changed Question #2 K/A and Description on JPM 1-#10 to reflect new question written.

Changed Question #2 K/A and Description on JPM 1-#5 to reflect new question written.

SRO-U Outline - Changed JPM 1-#3 from "Transferring From SDC To LPCI On Low Water Level" to "Transferring Operating RHR Pumps While In Shutdown Cooling". Also changed system tested from RHR-LPCI Mode to Shutdown Cooling.

Changed Question #2 K/A and Description on JPM 1-#10 to reflect new question written.

Changed Question #2 K/A and Description on JPM 1-#5 to reflect new question written.

JPM 1-#1 - Inserted actual control rods for JPM 18-11 for XX-YY and 18-15 for AA-BB.

Annotated Question #1 to allow use of prints as the only reference. Annotated Question #2 as "Closed Reference" only. Added JPM Question Handouts.

Modified Steps 3 and 10 to select the control rod prior to inserting it. Added Note to Step 15 that Candidate may withdraw rod from "44" to "48" and do the coupling check in one motion or step effectively combining Steps 14, 15 and 16.

Removed Critical Step designation from Step 14. Added Note to Step 13 that this rod is not a problem rod. Not in CRC Book. Added one Task Condition that a second operator is available for the rod verifications. Added note for Simulator operator that the third rod (18-19) can be used for the double notch if 18-15 is missed for any reason.

JPM 1-#2 - Fixed procedure typos in Step 1, should have been OP-152 vice ON-152. Fixed typo in Step 24 Action, 5000 gpm vs 2500 gpm. Removed the actual closure of F042 and F100 in Step 27 and made them verifications as they will already be closed. Also, simulator will not prevent HPCI turbine from tripping on this, so made that a "verify" in Step 27. Fixed Expected Answer indentation in Question #2.

Corrected typo on Step #12, made it Critical. Added JPM Question Handouts.

Modified JPM to reflect Rev. 25 to the procedure dated 04/21/99. No changes made to the actual JPM steps, just modified the references.

JPM 1-#3 - New JPM. Old 1-#3 JPM as developed pointed out a plant procedure problem for this task. The problem was such that with Unit 2 currently using SDC, the procedure change was mandatory and over-rode the need of the task for the JPM. New JPM written to transfer operating RHR Pump while in Shutdown Cooling including starting ESW.

Added JPM Question Handouts.

Added sentence to Initiating Cue that the simulator indication may not reflect recent procedure changes and that the evaluator will provide specific indication as required (at Step 9).

JPM 1-#4 - Fixed grammatical error with Question #1 Expected Response.

Annotated Question #1 as "Closed Reference" only. Rewrote Question #1 to ask for indications of uncovered RHR Pump suction as well as adverse plant response to this failure. Added JPM Question Handouts.

Added the words "or pump damage" to Question #1 to preclude that being given as an answer.

JPM 1-#5 - Step 5 Sync Selector Switch key not kept in locker. It is in Tie Breaker switch on the panel. Added the option to open the DG output breaker to correct overload problem in Step 13. Either or both actions are correct and are critical.

Corrected typo on Step #8, made it Critical. Annotated Question #1 as "Closed Reference" only. Increased level of difficulty on Question #2 by asking status of engine from a Control Room stop signal, how to immediately stop the engine and then a print exercise to prove the answer. Annotated Question #2 to allow use of prints as the only reference. Added JPM Question Handouts.

Wrote new Question #2 on the "E" DG and ESW response to LOOP after review of original question indicated the prints required to answer it were difficult to read/use. New question also requires prints to answer.

JPM 1-#6 - Typo or incorrect answer on Question #2. ADS SRVs will remain open if ECCS Pumps are secured.

Reworded Question #1 to make easier to ask in oral exam format. Annotated Question #1 to allow use of Tech Specs as only reference. Added JPM Question Handouts.

<u>Modified Task Conditions and Initiating Cue to direct the Candidate to respond to alarms.</u>

JPM 1-#7 - Fixed valve number typo in Step 5.

NRC comment on Question #1 was that it is same as #56 on the written exam. That question is testing knowledge of Fire Suppression relationship to SGTS Operability concerns. This JPM question is testing the relationship between SGTS and Secondary Containment therefore no changes were made at this time. Added JPM Question Handouts.

Fixed grammatical error in Step 5, removed the "the". **Removed the unnecessary ")" in Step 10.**

Added Task Condition that Unit 2 is not venting their drywell. Added note to Evaluator that Candidate may refer to OP-173-003 first which directs use of OP-070-001.

JPM 1-#8 - Added "SLC" before the Squib Valves in Question #2.

Added JPM Question Handouts.

Fixed some Step Action and Standard alignment problems.

JPM 1-#9 - **Added JPM Question Handouts.**

Rewrote the JPM Task Conditions to be less specific on the current plant conditions and Initiating Cue to be more specific on the exact procedure section to be performed. This conforms more with how it would actually be done in a real emergency.

JPM 1-#10 - **Annotated Question #1 as "Closed Reference" only. Wrote new Question #2 asking how to withdraw a single rod with consecutive RPIS reed switch failures. Added JPM Question Handouts.**

Changed power level in Question #1 to 8% vice 15% to reflect the RSCS/RWM changes made with Improved Tech Specs. Updated JPM to reflect Rev. 9 of OP-156-002 dated 03/08/99. No major changes made to JPM.

Added actual arrangement of the toggle switches as found on panel to Step 7.

TN 03)

PENNSYLVANIA POWER & LIGHT COMPANY
JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC 1-#1	0	05/10/99	201002	1
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Take Actions For A Control Rod Double Notch

Completed By:

C. J. Tyner ____ Writer 03/10/99 ____ Date

Reviews:

____ Instructor/Writer ____ Date

Approval:

____ Requesting Supv./C.A. Head ____ Date ____ Nuclear Training Supv. ____ Date

---

ALTERNATE PATH: NO TIME CRITICAL: NO LOW POWER/SHUTDOWN: NO

TESTING METHOD: PERFORM - SIMULATOR

JPM SOURCE: NEW

Date of Performance:

____ 10 Min Allowed Time (Min) ____ Time Taken (Min)

JPM Performed By:

____ Last First M.I. ____ Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Ques #1: ( ) Satisfactory ( ) Unsatisfactory Ques #2: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

____ Signature ____ Typed or Printed

Comments:

REQUIRED TASK INFORMATION
JOB PERFORMANCE MEASURE
NRC 1-#1

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. ON-155-001, "Control Rod Problems", Rev. 14, Section 3.7

B. SO-156-001, "Weekly Control Rod Exercising", Rev. 11

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. Unit 1 is operating at power

B. SO-156-001, "Weekly Control Rod Exercising", is in progress

C. Control rod XX-YY is the next control rod to be exercised

V. INITIATING CUE

The Unit Supervisor directs you to continue with and complete SO-156-001, starting with control rod XX-YY

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#1

Candidate Name:______________________

Step	Action	Standard	Eval	Comments
	Simulator Setup • Provide copy of SO-156-001 and Attachment A. Markup through rod XX-YY. • Any at-power IC • Start the SO with a fully withdrawn control rod, and insert malfunction for a double notch on the second fully withdrawn rod to be inserted Only want the rod to go in 2 notches • Allow the rod to be recovered and withdrawn back to Notch "48"			
1.	Obtain and review controlled copy of SO-156-001 **Evaluator** - Candidate may review previous SO-156-001 steps and rods	Controlled copy of SO-156-001 obtained, reviews prerequisites and precautions		
2.	Selects applicable procedure section at control rod XX-YY	Selects Section 6.5 for rod XX-YY		
*3.	Insert control rod XX-YY one notch	Presses the Insert pushbutton momentarily		
4.	Monitors for proper response	Checks control rod position indication for rod insertion, may monitor CRD system parameters, reactor power, etc.		
*5.	Withdraw control rod XX-YY one notch **Evaluator** - Candidate may use Continuous Withdraw here to perform coupling check per 6.5.7	Presses the Withdraw pushbutton momentarily		

* - Critical Step   # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#1

Candidate Name:________________

Step	Action	Standard	Eval	Comments
6.	Monitors for proper response	Checks control rod position indication for rod return to position on Data Sheet, may monitor CRD system parameters, reactor power, etc.		
7.	Confirm completion of operability check	Circles "Sat" in Operability Check column of Data Sheet, no problems with rod to record		
*8.	Perform coupling check for control rod XX-YY	Presses Withdraw pushbutton and checks the following: • Records Drive Water Flow in Withdrawal Stall Flow column • Notch "48" indicated • Full-Out red light on full core display • No Rod Overtravel received • Circles "Sat" in Full Out Position Indication Check column		
9.	Confirm completed actions on rod XX-YY **Evaluator** - another operator will initial the Verify column	Initials Confirm column		
*10.	Insert control rod AA-BB one notch	Presses the Insert pushbutton momentarily		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#1

Candidate Name:____________________

Step	Action	Standard	Eval	Comments
*11.	Monitors for proper response, recognize rod double notches to Notch "44" **Evaluator** - acknowledge double notch as US	Checks control rod position indication for rod insertion, recognizes rod continues in to Notch "44", informs US		
*12.	Enters ON-155-001 **Evaluator** - acknowledge ON-155-001 entry as US	Enters and takes actions IAW Section 3.7 of ON-155-001, informs US		
13.	Documents double notch of rod AA-BB	Completes Attachment A of ON-155-001 for rod AA-BB		
*14.	Withdraw rod AA-BB to Notch "46"	Presses the Withdraw pushbutton, observes normal parameters, notes rod moves out to Notch "46", informs US		
*15.	Withdraw rod AA-BB to Notch "48" **Evaluator** - Candidate may use Continuous Withdraw here to perform coupling check per 6.5.7	Presses the Withdraw pushbutton, observes normal parameters, notes rod moves out to Notch "48", informs US, records data on Attachment A		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#1

Candidate Name:____________________

Step	Action	Standard	Eval	Comments
*16.	Perform coupling check for control rod AA-BB	Presses Withdraw pushbutton and checks the following: • Records Drive Water Flow in Withdrawal Stall Flow column • Notch "48" indicated • Full-Out red light on full core display • No Rod Overtravel received • Circles "Sat" in Full Out Position Indication Check column		
17.	Confirm completed actions on rod AA-BB **Evaluator** - another operator will initial the Verify column **Evaluator** - Another operator will continue the control rod exercising	Initials Confirm column		

* - Critical Step # - Critical Sequence

TASK CONDITIONS:

A. Unit 1 is operating at power

B. SO-156-001, "Weekly Control Rod Exercising", is in progress

C. Control rod XX-YY is the next control rod to be exercised

INITIATING CUE

The Unit Supervisor directs you to continue with and complete SO-156-001, starting with control rod XX-YY

TASK CONDITIONS:

A. Unit 1 is operating at power

B. SO-156-001, "Weekly Control Rod Exercising", is in progress

C. Control rod XX-YY is the next control rod to be exercised

INITIATING CUE

The Unit Supervisor directs you to continue with and complete SO-156-001, starting with control rod XX-YY

Appl. To/JPM No: NRC 1-#1 Candidate Name:____________________

QUESTION NO: 1

With the plant at normal operating conditions, how does throttling the CRD Pressure Control Valve in the closed direction result in an increased drive water pressure (D/P)? How does the Flow Control Valve respond during this change of pressure? Explain your answer?

EXPECTED ANSWER:

- For a steady flowrate, closing the valve raises the differential pressure across the valve
- The FCV will open
- As the PCV is closed, flow though the system is reduced, therefore FCV opens to restore flow to setpoint

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 201001K407 3.1/3.0

REFERENCES: SY017 K-2, "Control Rod Drive Hydraulics", Rev. 2, Section IV.B.4.c.1)h), Page 16, LO - 7.c

Appl. To/JPM No: NRC 1-#1 Candidate Name:________________________

QUESTION NO: 2

How is the reactivity insertion rate (required control rod speed) regulated during a SINGLE control rod withdrawal/insertion movement? Though not a Tech Spec number, normal control rod movement speeds are limited. What is the basis for limiting the maximum speed for control rod movement?

EXPECTED ANSWER:

- The water flow going to and leaving from the under-piston area of the control rod drive mechanism is throttled. (needle valves on directional control valves 120 and 123)
- Maximum speed is based upon limiting the maximum reactivity addition rate during a continuous control rod withdrawal accident during a startup.

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 201001K110 2.8/2.8

REFERENCES: SY017 K-2, "Control Rod Drive Hydraulics", Rev. 1, Section IV.A.4.d, Pages 4 & 5, LO - 7.e

USAR, Volume 7, Section 4.6

7083 2

PENNSYLVANIA POWER & LIGHT COMPANY
JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC 1-#2	0	05/10/99	206000	2
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Place HPCI In CST To CST Mode - Steam Leak Without Isolation

Completed By: **Reviews:**

C. J. Tyner ________ 03/10/99 ________ ____________ ________
Writer Date Instructor/Writer Date

Approval:

____________ ________ ____________ ________
Requesting Supv./C.A. Head Date Nuclear Training Supv. Date

ALTERNATE PATH: YES TIME CRITICAL: NO LOW POWER/SHUTDOWN: NO

TESTING METHOD: PERFORM - SIMULATOR

JPM SOURCE: NEW

Date of Performance:

________ 10 Min ________
Allowed Time (Min) Time Taken (Min)

JPM Performed By:

____________________ ____________
Last First M.I. Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Ques #1: ( ) Satisfactory ( ) Unsatisfactory Ques #2: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

____________ ____________
Signature Typed or Printed

Comments:

REQUIRED TASK INFORMATION
JOB PERFORMANCE MEASURE
NRC 1-#2

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. OP-152-001, "High Pressure Coolant Injection", Rev. 24, Section 3.2.9

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. Unit 1 is operating at power

B. Maintenance needs to take vibrations readings on HPCI

C. Suppression pool cooling is in service

D. Standby Gas Treatment and ESW are in service

E. An operator is standing by in the HPCI Room and HPCI has been verified filled and vented.

V. INITIATING CUE

The Unit Supervisor directs you to perform a manual start of HPCI and place it in the CST to CST Mode at 5000 gpm at 900 psig discharge pressure with the flow controller in "Automatic" in accordance with OP-152-001, Section 3.2.9. A second operator is available to perform SO-159-010.

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#2 Candidate Name:________________

Step	Action	Standard	Eval	Comments
0.	**Simulator Setup:** • Any at-power IC • Place suppression pool cooling, Standby Gas Treatment and ESW in service • Support Candidate as requested for HPCI operation • Allow candidate to start HPCI and place in "Automatic" at 5000 gpm then insert malfunction for steam leak with a failure to auto isolate. Fail the Manual Isolation as well. Allow the valves to close when Candidate closes them with the switches			
1.	Obtain controlled copy of ON-152-001	Controlled copy of ON-152-001 obtained.		
2.	Selects applicable procedure section	Selects Section 3.2.9		
3.	Review prerequisites and precautions	Ensure prerequisites and precautions are met		
4.	Place HPCI Div 1 and 2 MOV OL Bypass keyswitches to "Test"	HS-E41-1S42 and 1S41 in "Test", HPCI Out Of Service annunciator received, HPCI Div 1 and 2 MOVS In Test lights on		
5.	Place HPCI Div 1 and 2 Out Of Service switches to "Inop"	HS-E41-1S34A and 1S34B in "Inop", HPCI Div 1 and 2 Out Of Service status lights on		
6.	Ensure HPCI Pump suction pressure is GTE 18 psig	Checks PI-E41-1R606 GTE 18 psig		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#2

Candidate Name:______________________

Step	Action	Standard	Eval	Comments
7.	Perform SO-159-010 __Evaluator__ - the second operator is ready to perform SO-159-010	Identifies requirement to perform SO-159-010		
8.	Ensure HPCI Injection Valve (F006) is closed	Checks HV-155-F006 closed		
9.	Open Breaker 1D264061 for HPCI Injection Valve __SIM OP__ - open breaker when Candidate requests	Directs local operator to open breaker. Verifies HPCI Div 2 OL or Power Loss light is on when breaker is opened		
10.	Place SGTS, ESW and Supp Pool cooling in service	Verifies all in service, from initial conditions		
11.	Check HPCI Test Line to CST Valve (F008) closed	Verifies HV-155-F008 closed		
12.	Open HPCI Test Line to CST Valve (F011)	Opens HV-155-F011		
13.	Check HPCI filled and vented	Verifies HPCI filled and vented, from initial conditions		
14.	Evacuate personnel from HPCI Room and Pipe areas	Makes announcement to evacuate areas/directs local operator to leave area		
15.	Place HPCI Flow Controller in "Manual" and set at "Minimum"	Places FC-E41-1R600 in "Manual" and runs tape down to "Minimum"		
16.	Start HPCI Barometric Condenser Vacuum Pump	Starts 1P216		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#2

Candidate Name:_______________

Step	Action	Standard	Eval	Comments
17.	Open HPCI Lube Oil Cooling Water Valve (F059)	Opens HV-156-F059		
*18.	Simultaneously start HPCI Aux Oil Pump and open Turbine Steam Supply Valve (F001)	Starts 1P213 and opens HV-155-F001		
19.	Ensures normal HPCI startup response	Ensures the following: • HPCI Line Drain To Condenser Inboard and Outboard Isolation valves (HV-155-F028 & F029) close • HPCI Barometric Condenser Condensate Pump Discharge Drain Valves (HV-155-F025 & F026) close if open • HPCI Room Cooler (1V209A/B) starts at 1C681 • HPCI Min Flow To Suppression Pool Valve (HV-155-F012) opens • HPCI Pump Discharge Low Flow alarm is received after time delay • Full open indication on HPCI Turbine Stop Valve (FV-15612)		
*20.	Raise HPCI turbine speed to approx 2200-2500 rpm using HPCI Flow Controller (FC-E41-1R600)	HPCI turbine speed raised to between 2200-2500 with flow controller in "Manual"		
21.	Throttle open HPCI Test Line to CST Isolation Valve (F008) and adjust HPCI Flow Controller (FC-E41-1R600) to achieve approx 2500 gpm	Throttles open HV-155-F008 and adjusts turbine speed to obtain approx 2500 gpm on FI-E41-1R600-1		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#2

Candidate Name:_______________

Step	Action	Standard	Eval	Comments
22.	Ensure normal HPCI response	Ensures the following occur: • HPCI Aux Oil Pump (1P213) stops • HPCI Min Flow To Suppression Pool Valve (HV-155-F012) closes • HPCI Pump Discharge Low Flow alarm clears		
23.	Null the HPCI Flow Controller and place in "Automatic"	Nulls the HPCI Flow Controller (FC-E41-1R600) and places in "Automatic"		
*24.	Adjusts HPCI Test Line to CST Isolation Valve (F008) and HPCI Flow Controller (FC-E41-1R600) to achieve approx 2500 gpm	HV-155-F008 and turbine speed adjusted to obtain approx 5000 gpm on FI-E41-1R600-1 at 900 psig discharge pressure		
*25.	Recognize/take action for AR-114-001 A02, A03, F04 & F05 inform US **Evaluator** - Unit Supervisor acknowledges	Acknowledges and silences alarms, informs US		
*26.	Recognize HPCI steam leak indications and failure to isolate, inform US **Evaluator** - Unit Supervisor acknowledges	Recognizes steam leak indications and a HPCI failure to isolate, informs US		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#2

Candidate Name:____________________

Step	Action	Standard	Eval	Comments
*27.	Isolate HPCI and trip HPCI turbine **Evaluator** - Unit Supervisor acknowledges, Candidate should NOT have to refer to AR-114-001 for required actions.	Isolates and trips HPCI: • Closes Steam Supply Inboard & Outboard Isolation Valves (HV-155-F002 & F003) • Closes Warm Up Line Isolation Valve (HV-155-F100) • Closes Pump Suction From Suppression Pool (HV-155-F042) • Trips HPCI turbine • Informs US HPCI isolation actions completed		
28.	Check HPCI Leak Detection for indications of steam line break **Evaluator** - Inform Candidate another operator will complete the remaining actions for securing HPCI	Checks for steam line break indications		

* - Critical Step # - Critical Sequence

TASK CONDITIONS:

A. Unit 1 is operating at power

B. Maintenance needs to take vibrations readings on HPCI

C. Suppression pool cooling is in service

D. Standby Gas Treatment and ESW are in service

E. An operator is standing by in the HPCI Room and HPCI has been verified filled and vented.

INITIATING CUE:

The Unit Supervisor directs you to perform a manual start of HPCI and place it in the CST to CST Mode at 5000 gpm at 900 psig discharge pressure with the flow controller in "Automatic" in accordance with OP-152-001, Section 3.2.9. A second operator is available to perform SO-159-010.

TASK CONDITIONS:

A. Unit 1 is operating at power

B. Maintenance needs to take vibrations readings on HPCI

C. Suppression pool cooling is in service

D. Standby Gas Treatment and ESW are in service

E. An operator is standing by in the HPCI Room and HPCI has been verified filled and vented.

INITIATING CUE:

The Unit Supervisor directs you to perform a manual start of HPCI and place it in the CST to CST Mode at 5000 gpm at 900 psig discharge pressure with the flow controller in "Automatic" in accordance with OP-152-001, Section 3.2.9. A second operator is available to perform SO-159-010.

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#2 Candidate Name:____________________

QUESTION NO: 1

HPCI is running and injecting to the reactor following a valid initiation signal. The flow controller is in "Automatic" with flow at 5000 gpm. What would be the effect on HPCI if the Ramp Generator failed to its "low" limit? Explain your answer. Can flow be restored to 5000 gpm by placing the flow controller in "Manual"? Explain your answer.

EXPECTED ANSWER:

- Turbine speed and pump flow lower
- The flow signal and the ramp generator signal both input into a "low" signal selector which passes the lowest signal on to the speed control circuitry. The "low" signal is now the output signal from the failed ramp generator so the turbine receives a speed reduction signal.
- No
- The manual signal from the controller goes to the "low" signal selector same as the automatic signal

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 206000K505 3.3/3.3

REFERENCES: SY017 C-6, "High Pressure Coolant Injection System", Rev. 2, Figure 18, LO - 8.d, 9 & 10

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#2 Candidate Name:____________________

QUESTION NO: 2

Describe how and why HPCI operating in the CST to CST mode for surveillance testing will affect steady state plant operation at maximum power. What actions are directed to prevent these problems?

EXPECTED ANSWER:

– Running HPCI will cause a small loss of feedwater heating due to HPCI utilizing some of the steam that had been going to feed heating. This causes a rise in reactor power. (Approx 15 MWe)
– While performing this surveillance, the operator is directed to maintain reactor power LTE 100%

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 206000A217 3.9/4.3

REFERENCES: SO-152-002, "Quarterly HPCI Flow Verification", Rev. 24, Section 5.17, Page 7

SY017 C-6, "High Pressure Coolant Injection System", Rev. 2, Section II.C, Page 1, LO - 9

F-?? 3

PENNSYLVANIA POWER & LIGHT COMPANY
JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC 1-#3	0	05/10/99	205000	4
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Transfer from Shutdown Cooling to LPCI Operation Due to Low RPV Level (Level 1)

Completed By: Reviews:

C. J. Tyner	03/10/99		
Writer	Date	Instructor/Writer	Date

Approval:

Requesting Supv./C.A. Head	Date	Nuclear Training Supv.	Date

ALTERNATE PATH: NO TIME CRITICAL: NO LOW POWER/SHUTDOWN: YES

TESTING METHOD: PERFORM - SIMULATOR

JPM SOURCE: Facility JPM 49.OP.020.102, Rev. 0 - Modified to actually inject with the new lineup, updated JPM to latest procedure revisions

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Date of Performance:

	20 Min	
	Allowed Time (Min)	Time Taken (Min)

JPM Performed By

Last	First	M.I.	Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Ques #1: ( ) Satisfactory ( ) Unsatisfactory Ques #2: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

Signature	Typed or Printed

Comments:

REQUIRED TASK INFORMATION
JOB PERFORMANCE MEASURE
NRC 1-#3

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. OP-149-002, "RHR Operation in Shutdown Cooling Mode", Rev. 26, Section 3.7

B. OP-149-001, "RHR System", Rev. 23, Section 3.2

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. The Plant is shut down and shutdown cooling is in service, using RHR Pump 1P202A.

B. A LPCI initiation signal due to low RPV level (Level 1) has been received.

C. The "B" RHR Loop and Core Spray systems are not available.

V. INITIATING CUE

The Unit Supervisor directs you to transfer RHR Loop "A" from shutdown cooling to LPCI mode and inject to the reactor to raise reactor water level

PERFORMANCE CHECKLIST

Appl. To/JPM No.: NRC 1-#3

Candidate Name:________________

Step	Action	Standard	Eval	Comments
1.	Obtain a controlled copy of OP-149-002.	Controlled copy obtained.		
2.	Select the correct section to perform.	Selects Section 3.7.		
3.	Review the prerequisites. **Evaluator** Inform Candidate all prerequisites have been met.	Ensures that all prerequisites have been met.		
4.	Go to Step 3.7.4.	Selects Step 3.7.4.		
*5.	Ensure that RHR Loop A pump has tripped. Pump will not be tripped, it will be cycling. Must take switches to stop.	Checks that the following have tripped by placing HS to stop: • RHR Pump 1P202A • RHR Pump 1P202C		
6.	Observe the White "Override" lights lit for the "A" and "C" RHR Pumps	Checks "Override" lights on for the "A" and "C" RHR Pumps		
7.	Close the shutdown cooling suction valves.	Places the control switches for the following valves in the CLOSE position: • Shutdown Clg Suc HV-151-F006A • Shutdown Clg Suc HV-151-F006C		
8.	Check the position of the RHR crosstie valves.	Confirms that the following valves are closed: • RHR Loop A Crosstie HV-151-F010A • RHR Loop B Crosstie HV-151-F010B		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No.: NRC 1-#3

Candidate Name:______________________

Step	Action	Standard	Eval	Comments
9.	Depressurize RHR Pumps "A" and "C" suction piping.	Depressurizes RHR Pump 1P202A & C suction piping by: • Places Keylock Switch 1S62A to test. • Places the RHR PP A/C Min Flow HV-151-F007A in the OPEN position. • Return Keylock Switch 1S62A to Normal.		
10.	Open the RHR Loop A Pump Suction Valves (F004A & 4C)	Places the control switches for the following valves in the OPEN position: • RHR Pump A Suct HV-151-F004A • RHR Pump C Suct HV-151-F004C		
*11.	Reset the injection valve logic.	Presses the following pushbuttons: • RHR Loop A Shutdown Clg Reset HS-E11-1S32A • RHR Loop B Shutdown Clg Reset HS-E11-1S32B		
12.	Observe that the Outboard Injection Valve (F015A) opens	Notes that RHR Inj OB Iso HV-151-F015A opens.		
*13.	Close the Inboard Injection Valve (F017A)	Places control switch for Inj Flow Ctl HV-151-F017A to Close.		
*14.	Start at least one RHR pump.	Places the control switch(es) for the following in the START position: • RHR Pump 1P202A • RHR Pump 1P202C		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No.: NRC 1-#3

Candidate Name:________________________

Step	Action	Standard	Eval	Comments
15.	Go to OP-149-001 to inject **Evaluator** - If Candidate asks, another operator will verify the non-RHR Section 3.2 actions for a LPCI initiation signal	Transitions to OP-149-001, Section 3.2		
*16.	Open the Inboard Injection Valve (F017A)	Places control switch for Inj Flow Ctl HV-151-F017A to Open		
17.	Verifies RHR Pump Minimum Flow Valve (F007A) closes	Checks HV-151-F007A closes at approx 3000 gpm		
18.	Verify reactor water level rising	Monitors reactor water level, informs US when level rising.		

* - Critical Step # - Critical Sequence

TASK CONDITIONS

A. The Plant is shut down and shutdown cooling is in service using RHR Pump 1P202A.

B. A LPCI initiation signal due to low RPV level (Level 1) has been received.

C. The "B" RHR Loop and Core Spray systems are not available.

INITIATING CUE

The Unit Supervisor directs you to transfer RHR Loop "A" from shutdown cooling to LPCI mode and inject to the reactor to raise reactor water level

TASK CONDITIONS

A. The Plant is shut down and shutdown cooling is in service using RHR Pump 1P202A.

B. A LPCI initiation signal due to low RPV level (Level 1) has been received.

C. The "B" RHR Loop and Core Spray systems are not available.

INITIATING CUE

The Unit Supervisor directs you to transfer RHR Loop "A" from shutdown cooling to LPCI mode and inject to the reactor to raise reactor water level

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#3 Candidate Name:______________________

QUESTION NO: _1_

With Unit 1 in Mode 4, a total loss of shutdown cooling occurs. Temperature is rapidly rising. One primary airlock door has been removed and sent offsite for repairs. When are actions required to be taken for these conditions? What actions are required to be taken.

EXPECTED ANSWER:

- No action REQUIRED until Mode 3 entered (>200 degrees F) but actions below will probably be taken.
- Verify the operable airlock door is closed within 1 hour and lock the operable airlock door closed within 24 hours. Verify the locked door to be locked at least once per 31 days. Otherwise be in Mode 3 (already there) within 12 hours and back in Mode 4 within 36 hours.

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 295021K201 3.6/3.7

REFERENCES: Unit 1 Tech Spec 3.6.1.2, Page 3.6-4

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#3 Candidate Name:____________________

QUESTION NO: 2

Unit 1 reactor temperature is 225 degrees F cooling down with Shutdown Cooling in operation. It has been determined that PIS-B31-1NO18A, is inoperable based upon a review of a previous calibration.

What actions are required?

EXPECTED ANSWER:

Table 3.3.6.1-1, Page 6 of 6, requires 1 instrument per trip system. The design has only 1 instrument per trip system. The inoperable channel be placed in "trip" within 24 hours and ensure the affected penetration flowpath (SDC) is isolated within one hour.

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 205000G222 3.4/4.1

REFERENCES: Unit 1 Tech Spec 3.3.6.1, Page 3.3-52

TDS 4

PENNSYLVANIA POWER & LIGHT COMPANY
JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC 1-#4	0	05/10/99	219000	5
Appl To	**JPM Number**	**Rev No.**	**Date**	**NUREG 1123 Sys. No.**	**SFG**

Task Title: Place RHR in Suppression Pool Cooling At The Remote Shutdown Panel

Completed By:

Reviews:

C. J. Tyner — Writer — 03/10/99 Date

________ Instructor/Writer ________ Date

Approval:

________ Requesting Supv./C.A. Head ________ Date ________ Nuclear Training Supv. ________ Date

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ALTERNATE PATH: NO TIME CRITICAL: NO LOW POWER/SHUTDOWN: NO

TESTING METHOD: PERFORM - SIMULATOR

JPM SOURCE: Facility JPM 49.OP.008.101, Rev. 0 - direct from source, updated to latest procedure revisions

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Date of Performance:

________ 20 Min Allowed Time (Min) ________ Time Taken (Min)

JPM Performed By

________ Last First M.I. ________ Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Ques #1: ( ) Satisfactory ( ) Unsatisfactory Ques #2: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

________ Signature ________ Typed or Printed

Comments:

REQUIRED TASK INFORMATION
JOB PERFORMANCE MEASURE
NRC 1-#4

I. **SAFETY CONSIDERATIONS**

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. **REFERENCES**

A. ON-100-009, "Control Room Evacuation", Rev. 4, Section 4.3
B. OP-149-005, "RHR Operation in Suppression Pool Cooling Mode", Rev. 17, Section 3.5

III. **REACTIVITY MANIPULATIONS**

N/A

IV. **TASK CONDITIONS**

A. A condition has occurred requiring abandonment of the Control Room.

B. All required immediate operator actions of ON-100-009 have been completed prior to abandoning the Control Room.

C. Transfer switch positions have been changed on the RSP IAW ON-100-009, Section 4.3.

D. Reactor pressure is being maintained by the SRVs cycling.

E. RPV water level is >-38 inches and stable.

F. ESW System is in service IAW OP-054-001.

G. RHRSW "B" Loop is in service IAW OP-116-001.

V. **INITIATING CUE**

The Unit Supervisor directs you to place RHR "B" Loop in Suppression Pool Cooling.

PERFORMANCE CHECKLIST

Appl. To/JPM No.: NRC 1-#4 Student Name:____________

Step	Action	Standard	Eval	Comments
	Simulator Setup: • Establish RPV water level approximately "0" inches • Complete operator actions for Control Room Evacuation in the IAW ON-100-009. • If NOT performing JPM 00.ON.015.101 prior to this JPM, transfer Control and Instrumentation to the RSDP IAW ON-100-009. • Start B and D ESW pumps. • Place "B" Loop RHRSW in service at 9,000 gpm. • Place Simulator in FREEZE. • When ready, place Simulator in RUN.			
1.	Obtain a controlled copy of OP-149-005. Evaluator - Student may review previous sections of ON-100-009.	Controlled copy of OP-149-005 obtained.		
2.	Select correct section(s) to perform.	Selects Section 3.5.		
3.	Review prerequisites.	Ensure prerequisites are met.		
4.	Review precautions when controlled from RSDP: • RHR MIN FLOW 1F007B will not auto open or close. • RHR Pump B will not auto start on LPCI signal. • RHR Loop "B" will not auto align for LPCI.	Follows precautions while performing RHR operations.		
5.	Ensure ESW Loop "B" and RHRSW in operation	Operating per turnover.		
6.	Stop the "2B" RHR Pump if running. Evaluator - When "2B" RHR Pump is addressed, inform Candidate, "2B" RHR Pump is not running.	Directs Local Operator to verify "2B" RHR Pump not running		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No.: NRC 1-#4

Student Name:______________________

Step	Action	Standard	Eval	Comments
*7.	Align RHR Loop "B" for a pump start.	Aligns RHR Loop "B" as follows: • Checks HX B SHELL-SIDE BYPASS HV-151-F048B open. • Opens SUPPRESSION CHAMBER SPRAY TEST SHUTOFF HV-151-F028B. • Check RHR MIN FLOW HV-151-F007B open.		
8.	Ensure "B" Loop RHR is filled and vented. Evaluator - When requested, inform Candidate "B" Loop RHR local discharge pressure is 75 psig. Evaluator - When requested as NLO, inform Candidate "B" Loop RHR has been manually checked filled and vented IAW OP-149-001, Section 3.6 and "B" Loop RHR Pumps are checked ready for a start.	Directs NLO to obtain "B" Loop RHR local discharge pressure and to check RHR Loop "B" filled and vented.		
*9.	Start "B" RHR Pump.	Momentarily places handswitch for "B" RHR Pump 1P202B to START.		
*10.	Establish flow to suppression pool.	• Throttles TEST LINE CTL HV-151-F024B to achieve and maintain flow through the heat exchanger, not to exceed 10,000 gpm. • Closes RHR Pump MIN FLOW HV-151-F007B when at least 3,000 gpm loop flow is reached. • Throttle closed HX B SHELL-SIDE BYPASS HV-151-F048B.		

* - Critical Step # - Critical Sequence

Appl. To/JPM No.: NRC 1-#4

Student Name:______________________________

Step	Action	Standard	Eval	Comments
11.	Ensure room cooler running. **<u>Evaluator</u>** When requested, inform the student RHR Room Cooler 1V202B is running.	Directs NLO to check RHR Room Cooler 1V202B running.		

* - Critical Step # - Critical Sequence

TASK CONDITIONS

A. A condition has occurred which requires abandonment of the Control Room.

B. All required immediate operator actions of ON-100-009, Control Room Evacuation, have been completed prior to abandoning the Control Room.

C. Transfer switch positions have been changed on the RSP IAW ON-100-009, Section 4.3.

D. Reactor pressure is being maintained by the SRVs cycling.

E. RPV water level is >-38 inches and stable.

F. ESW System is in service IAW OP-054-001.

G. RHRSW "B" Loop is in service IAW OP-116-001.

INITIATING CUE

The Unit Supervisor directs you to place RHR "B" Loop in Suppression Pool Cooling.

TASK CONDITIONS

A. A condition has occurred which requires abandonment of the Control Room.

B. All required immediate operator actions of ON-100-009, Control Room Evacuation, have been completed prior to abandoning the Control Room.

C. Transfer switch positions have been changed on the RSP IAW ON-100-009, Section 4.3.

D. Reactor pressure is being maintained by the SRVs cycling.

E. RPV water level is >-38 inches and stable.

F. ESW System is in service IAW OP-054-001.

G. RHRSW "B" Loop is in service IAW OP-116-001.

INITIATING CUE

The Unit Supervisor directs you to place RHR "B" Loop in Suppression Pool Cooling.

Appl. To/JPM No: NRC 1-#4 Student Name:________________________

QUESTION NO: 1

Assuming suppression pool temperature was 135 degrees F and rising and level was 19.5 feet and lowering prior to starting suppression pool cooling, what additional restrictions would be placed upon RHR Pump operation for these conditions?

EXPECTED ANSWER:

Pump operation may result in equipment damage for pool levels when level lowers to less than 18 feet.

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 219000A201 3.0/3.1

REFERENCES: EO-100-103, "Primary Containment Control", Figure 7 VL

Appl. To/JPM No: NRC 1-#4 Student Name:______________________

QUESTION NO: 2

If the "2B" RHR Pump had been left running and the "1B" RHR Pump had been started from the Remote Shutdown Panel, what would have been the result? Prove your answer utilizing the appropriate prints.

EXPECTED ANSWER:

The "1B" RHR Pump would have NOT have started (the interlock with the Unit 2 "B" RHR Pump is not defeated when operating from the RSP.)

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 295016K201 4.4/4.5

REFERENCES: SY017 C-1, "Residual Heat Removal System", Rev. 2, Figure 10, LO - 7, 12.a and 19.a

RHR Pump Start Logic

T935

PENNSYLVANIA POWER & LIGHT COMPANY
JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC 1-#5	0	05/10/99	264000	6
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Synchronize Diesel Generator B To 4.16 KV Bus 2B - Runaway Diesel

Completed By: **Reviews:**

C. J. Tyner ______ 03/10/99 ______ ______________ ______
Writer Date Instructor/Writer Date

Approval:

______________ ______ ______________ ______
Requesting Supv./C.A. Head Date Nuclear Training Supv. Date

ALTERNATE PATH: YES TIME CRITICAL: NO LOW POWER/SHUTDOWN: NO

TESTING METHOD: PERFORM - SIMULATOR

JPM SOURCE: Facility JPM 24.OP.006.202, Rev. 0 - Modified to different diesel & alternate path, updated to latest procedure revisions

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Date of Performance:

______________ 20 Min ______________
Allowed Time (Min) Time Taken (Min)

JPM Performed By

______________ ______________
Last First M.I. Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Ques #1: ( ) Satisfactory ( ) Unsatisfactory Ques #2: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

______________ ______________
Signature Typed or Printed

Comments:

REQUIRED TASK INFORMATION
JOB PERFORMANCE MEASURE
NRC 1-#5

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.
2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

OP-024-001, "Diesel Generators", Rev. 33, Section 3.3

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. Diesel Generator "B" was started manually from OC653 in accordance with OP-024-001 and has been running unloaded for five minutes.

B. No other diesel generator is operating synchronized to the grid.

C. An NPO is standing by at the diesel.

V. INITIATING CUE

The Unit Supervisor directs you to manually synchronize Diesel Generator "B" with 4.16 KV Bus 2B and pick up 4,000 KW of load.

Step	Action	Standard	Eval	Comments
	<u>Note</u> Unless otherwise stated, all controls and indicators are located on Panel OC653. <u>Simulator Setup</u> - • Perform a manual start of the "B" Diesel Generator IAW OP-024-001 and allow it to stabilize • When Candidate closes output breaker and goes to "Raise" to pickup the initial 1000 KW load, fail the Speed Governor switch in the "Raise" position and allow DG to slowly ramp up to maximum load. • If taken to "Lower", the Speed Governor will not reduce DG load			
1.	Obtain a controlled copy of OP-024-001.	Controlled copy obtained.		
2.	Select the correct section to perform.	Selects Section 3.3 subsection 3.3.4		
3.	Review the prerequisites. <u>Evaluator</u> Inform the Candidate that all prerequisites have been met.	Ensures that all prerequisites have been met.		
4.	Review the precautions. <u>Evaluator</u> If asked, inform the Candidate that the diesel has been running for 15 minutes unloaded.	Follows the precautions as applicable.		

* - Critical Step # - Critical Sequence

Step	Action	Standard	Eval	Comments
5.	Obtain a key for the DG sync selector switch.	Obtains a key from the key locker.		
*6.	Turn the sync selector switch on. **Evaluator** When the switch is placed in the ON position the Synchroscope pointer will start moving (either direction), the white light on each side of the Synchroscope will flash off and on as the pointer rotates. The lights will be off when the pointer is between 10° before the 12 o'clock position and 10° after the 12 o'clock position.	Places the DG B to Bus 2B Sync Sel HS-00040B switch in the ON position.		
*7.	Adjust diesel generator voltage. **Evaluator** Voltage is matched when the pointer on the Diesel Gen Bus Diff Volts Meter is "0".	Takes the DG B Voltage Adjust HS-00053B switch to the RAISE or LOWER position as required to match Incoming and Running volts on the Diesel Gen Bus Diff Volts XI-00036 meter. (In green band)		
8.	Adjust diesel generator speed. **Evaluator** The FAST direction is clockwise.	Takes the DG B Speed Governor HS-00054B switch to the RAISE or LOWER position to cause the Synchroscope XI-00037 pointer to rotate slowly in the FAST direction. (≈1 rotation in one minute).		

* - Critical Step # - Critical Sequence

Step	Action	Standard	Eval	Comments
*9.	Close the diesel generator output breaker. **Evaluator** • Both white lights will be extinguished and the Synchroscope pointer will stop at the 12 o'clock position. • The Running Idle light will extinguish, and the Running Loaded light illuminates on the Local Panel OC521A.	Takes the DG B to Bus 2B Bkr 2A20204 switch to the CLOSE position when the Synchroscope XI-00037 pointer is at or slightly before the 12 o'clock position.		
*10.	Pick up load on the DG.	Immediately take and hold the DG B Speed Governor HS-00054B to the RAISE position until DG B Watts XI-00032A meter indicates ≥1,000 KW (over 30 to 45 seconds).		
*11.	Recognize DG B load continuing to rise after Speed Governor released, inform US **Evaluator** - Unit Supervisor acknowledges	Releases Speed Governor at approx 1000 KW, recognizes load continues to rise, informs US		
12.	Attempts to reduce DG B load **Evaluator** - Unit Supervisor acknowledges	Places DG B Speed Governor in to the LOWER position, recognizes load still rising, informs US		
*13.	Trip DG B	Press DG B STOP pushbutton,		

* - Critical Step # - Critical Sequence

Step	Action	Standard	Eval	Comments
14.	Verifies DG B no longer overloaded or is running in the cooldown mode, Inform US Evaluator - Unit Supervisor acknowledges. Inform Candidate that another operator will complete the remaining procedural steps and will initiate troubleshooting.	Checks DG B output breaker open, DG B running in cooldown mode, informs US		

* - Critical Step # - Critical Sequence

TASK CONDITIONS:

A. Diesel Generator A was started manually from OC653 in accordance with OP-024-001 and has been running unloaded for five minutes.

B. No other diesel generator is operating synchronized to the grid.

C. An NPO is stationed at the diesel.

INITIATING CUE:

The Unit Supervisor directs you to manually synchronize Diesel Generator "B" with 4.16 KV Bus 2B and pick up 4,000 KW of load.

TASK CONDITIONS:

A. Diesel Generator A was started manually from OC653 in accordance with OP-024-001 and has been running unloaded for five minutes.

B. No other diesel generator is operating synchronized to the grid.

C. An NPO is stationed at the diesel.

INITIATING CUE:

The Unit Supervisor directs you to manually synchronize Diesel Generator "B" with 4.16 KV Bus 2B and pick up 4,000 KW of load.

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#5 Candidate Name:________________________

QUESTION NO: 1

Given that the diesel generator is running after a manual start and is synched to its 4160 VAC bus, what will happen to the diesel if a LOCA occurs? What would occur if a loss of off-site power subsequently occurs?

EXPECTED ANSWER:

- The DG output breaker will open, engine control will swap to the isochronous mode.
- If the 4160 VAC bus is lost, the diesel will automatically re-synch to the bus. (output breaker will reclose)

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 264000A210 3.9/4.2

REFERENCES: SY017 G-1, "Diesel Generator", Rev. 3, Section IV.D.9.a.2), Page 25, LO - 11

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#5 Candidate Name:______________________

QUESTION NO: 2

For the conditions of this task how could the 5 minute cooldown be bypassed and the DG be immediately shutdown?

EXPECTED ANSWER:

The local operator could place the Mode Selector Switch to "Local" and press the Emergency Stop Pushbutton.

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 264000A404 3.7/3.7

REFERENCES: SY017 G-1, "Diesel Generators", Rev. 3, Fact Sheets Trips and Interlocks 3.5), Page 4

F99t

PENNSYLVANIA POWER & LIGHT COMPANY
JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC 1-#6	0	05/10/99	218000	3
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Respond To A Stuck Open Safety Relief Valve

Completed By: Reviews:

C. J. Tyner ____ 03/10/99 ____ ____ ____

Writer Date Instructor/Writer Date

Approval:

____ ____ ____ ____

Requesting Supv./C.A. Head Date Nuclear Training Supv. Date

ALTERNATE PATH: NO TIME CRITICAL: YES RCA ENTRY: NO

TESTING METHOD: PERFORM - SIMULATOR

JPM SOURCE: Facility JPM 83.ON.001.181, Rev. 0 - modified to different SRV, updated to current procedure revisions

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Date of Performance:

____ 5 Min ____

Allowed Time (Min) Time Taken (Min)

JPM Performed By

____ ____

Last First M.I. Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Ques #1: ( ) Satisfactory ( ) Unsatisfactory Ques #2: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

____ ____

Signature Typed or Printed

Comments:

REQUIRED TASK INFORMATION
JOB PERFORMANCE MEASURE
NRC 1-#6

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. ON-183-001, "Stuck Open Safety Relief Valve", Rev. 18, Section 3

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. Unit 1 is operating at 90% power.

B. A transient involving C601 Panel is about to occur.

C. You are the Extra PCO and are responsible for operating C601 controls.

V. INITIATING CUE

The Unit Supervisor directs you to perform all required operator actions to mitigate the consequences of the transient.

NOTE: Do NOT tell Candidate this is a Time Critical task.

Appl. To/JPM No.: NRC 1-#6

Candidate Name:______________________

Step	Action	Standard	Eval	Comments
	Evaluator **This is a TIME CRITICAL JPM and must be performed in the simulator.** • In order to successfully complete this JPM, the Candidate MUST give some indication that the reactor has to be scrammed within two minutes of the SRV opening. • The Candidate may obtain a copy of ON-183-001 at any time during the performance of the JPM, but still must meet the two minute time constraint. **Simulator Setup** • Select a 100 percent power IC (i.e., IC 18) and lower Recirc flow to result in 90% power. • Assign the following malfunction to a Function Button (Instructor Station or Hand-Held Remote): IMF RV01:PSV141F13B. This will cause the B SRV to inadvertently open and stay open. • When ready to begin, place the simulator to RUN. • When Candidate has read the Task Conditions/ Initiating Cue Sheet, DEPRESS the assigned Function Button to enter the malfunction.			

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No.: NRC 1-#6

Candidate Name:______________________

Step	Action	Standard	Eval	Comments
*1.	Recognize/take action for AR-110-E1, E2 & E-3, recognize SRV "B" is open, inform US/Control Room **<u>Evaluator</u>** - When/if Candidate states the time requirement, you (acting as US) should maintain the timeline and update the Candidate periodically as attempts are made to close the SRV. Do NOT provide the required action, just the elapsed time	• Determines that the "B" SRV is open • Announces open SRV to Control Room personnel. • States or gives indication that the event is time critical		
2.	Ensure SRV "B" should NOT be open	Verifies reactor pressure is: • Less than SRV lift setpoint • Less than 1,087 psig		
*3.	Attempt to close SRV "B"	Places the control switch for the "B" SRV to the OFF position.		
4.	Check for SRV closure <u>Cue</u> - Acknowledge report as US	Checks any of the following indications: • Acoustic monitor lights on 1C601or 1C690 extinguished • Tailpipe temperature decrease • Reactor thermal power or generator MWe increase • RPV pressure trend Recognizes SRV "B" NOT closed, informs US		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No.: NRC 1-#6

Candidate Name:______________________

Step	Action	Standard	Eval	Comments
*5.	Attempt to cycle SRV "B"	Places control switch for SRV "B" to the OPEN position, THEN to OFF position.		
6.	Checks for SRV closure. **Evaluator** The student may repeat the cycling sequence two or more times before continuing. As long as the two-minute time limit is not exceeded, this operation will not affect JPM performance evaluation.	Checks any of the following indications: • Acoustic monitor lights on 1C601or 1C690 extinguished • Tailpipe temperature decrease • Reactor thermal power or generator MWe increase • RPV pressure trend		
7..	Determine that the SRV has not closed. **Evaluator** - As US, inform Candidate when approximately one minute fifty seconds have elapsed, since the SRV opened	Announces/states SRV is still open.		
*8.	Scram the reactor/recommend reactor be scrammed. Note: This decision may be based upon SRV being open for nearly 2 minutes OR that the SRV cannot be closed. **Evaluator** - When Candidate has shown indication of the requirement to place the Mode Switch in SHUTDOWN, place the Simulator in FREEZE, and instruct the Candidate to stop	Recommends that the reactor be scrammed/places the Mode Switch in the SHUTDOWN position		

* - Critical Step # - Critical Sequence

TASK CONDITIONS

A. Unit 1 is operating at 90% power.

B. A transient involving C601 Panel is about to occur.

C. You are the Extra PCO and are responsible for operating C601 controls.

INITIATING CUE

The Unit Supervisor directs you to perform all required operator actions to mitigate the consequences of the transient.

TASK CONDITIONS

A. Unit 1 is operating at 90% power.

B. A transient involving C601 Panel is about to occur.

C. You are the Extra PCO and are responsible for operating C601 controls.

INITIATING CUE

The Unit Supervisor directs you to perform all required operator actions to mitigate the consequences of the transient.

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#6 **Candidate Name:**________________________

QUESTION NO: 1

With the plant operating at power, how would a failure of the "J" Safety Relief Valve bellows affect operation of the SRV and of the plant? How would the operator know the bellows has failed?

EXPECTED ANSWER:

- **Normal operation of the SRV in the ADS mode and the Relief Mode (Electrical/pneumatic operation) would be possible. The valve would NOT open in the Safety mode.**
- **Would not affect plant operation (must have the safety mode of 12 SRV operable)**
- **No indications until pressure reaches the safety setpoint of the valve. (Would not open at all or would open at a much higher pressure than required.)**

ACTUAL ANSWER:

SAT ________ **UNSAT** ________

K/A NUMBER: 218000K106 3.9/3.9

REFERENCES: SY017 C-4, "Automatic Depressurization And Overpressure Protection Systems", Rev. 1, Section III.A.2.c.(9) and Figures 2 & 3, Page 3, LO - 5

Unit 1 Tech Spec 3.4.3, Page 3.4-8

JPM QUESTIONS

Appl. To/JPM No: NRC 1-#6 Candidate Name:____________________

QUESTION NO: 2

Plant conditions resulted in an initiation of the Automatic Depressurization System. All ADS SRVs have opened.

What will be the expected ADS SRV response if all running low pressure ECCS pumps are secured? *Confirm your answer utilizing the appropriate logic prints.*

EXPECTED ANSWER:.

All ADS SRVs will close. (ADS logic is seal-in requiring operator action to reset to reclose the SRVs)

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 218000K601 3.9/4.1

REFERENCES: SY017 C-4, "Automatic Depressurization And Overpressure Protection Systems", Rev. 1, Section III.E.5 & Figure 5, Page 13, LO - 6 & 9

FA3 ?

PENNSYLVANIA POWER & LIGHT COMPANY
JOB PERFORMANCE MEASURE
APPROVAL AND ADMINISTRATIVE DATA SHEET

SRO	NRC 1-#7	0	05/10/99	261000	9
Appl To	JPM Number	Rev No.	Date	NUREG 1123 Sys. No.	SFG

Task Title: Perform a Manual Startup of the SGTS and Vent the Drywell

Completed By: Reviews:

C. J. Tyner — Writer; 03/10/99 — Date; _____ Instructor/Writer; _____ Date

Approval:

_____ Requesting Supv./C.A. Head; _____ Date; _____ Nuclear Training Supv.; _____ Date

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ALTERNATE PATH: NO TIME CRITICAL: NO LOW POWER/SHUTDOWN: NO

TESTING METHOD: PERFORM - SIMULATOR

JPM SOURCE: Facility JPM 70.OP.004.101, Rev. 0 - modified to vent the drywell, updated to latest procedure revisions

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Date of Performance:

_____ ; 10 Min — Allowed Time (Min); _____ Time Taken (Min)

JPM Performed By

_____ Last First M.I.; _____ Employee #/S.S. #

JPM Performance Evaluation: ( ) Satisfactory ( ) Unsatisfactory

Ques #1: ( ) Satisfactory ( ) Unsatisfactory Ques #2: ( ) Satisfactory ( ) Unsatisfactory

Evaluator Name:

_____ Signature; _____ Typed or Printed

Comments:

REQUIRED TASK INFORMATION
JOB PERFORMANCE MEASURE
NRC 1-#7

I. SAFETY CONSIDERATIONS

A. All Operations personnel are responsible for maintaining their radiation exposure As Low As Reasonably Achievable in accordance with OP-AD-001, Operations Shift Policies.

B. All applicable safety precautions shall be taken in accordance with established PP&L safety policies and the Safety Rule Book, for example:

1. Whenever any electrical panel is opened for inspection during JPM performance.

2. Whenever entering any plant area where specific safety equipment; such as hearing or eye protection, safety shoes, hardhats, etc; is required and/or posted as being necessary.

II. REFERENCES

A. OP-070-001, "Standby Gas Treatment System", Rev. 17, Section 3.2

B. OP-173-003, "Primary Containment Nitrogen Makeup and Venting", Rev. 5, Section 3.3

III. REACTIVITY MANIPULATIONS

N/A

IV. TASK CONDITIONS

A. The Unit 1 Drywell pressure has slowly been rising over the last 2 shifts

B. The SGTS is aligned for automatic initiation in accordance with OP-070-001.

C. All prerequisites, Tech Spec and TRM requirements have been met.

V. INITIATING CUE

The Unit Supervisor directs you to perform a manual startup of "A" Standby Gas Treatment train and vent the drywell to reduce pressure by 0.5 psig.

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#7

Candidate Name:____________________

Step	Action	Standard	Eval	Comments
	Simulator Setup: • Any at-power IC • Ensure the "A" SGTS Train lined up for normal auto start IAW OP-070-001 • Attempt to have a slightly elevated drywell pressure			
1.	Obtain a controlled copy of OP-070-001.	Controlled copy obtained.		
2.	Select the correct section to perform.	Selects Section 3.2.		
3.	Review the prerequisites: Evaluator - Inform Candidate that all prerequisites have been met.	Ensures that all prerequisites have been met.		
4.	Review the precautions.	Follows precautions as applicable.		
*5.	Open the SGTS outside the air damper. Note - HD-0755A remains open for 70 seconds after pushbutton released. Must establish flowpath before that time or it will close and SGTS Fan will not start.	Depress the OPEN pushbutton for SGTS Clg OA Dmp HD-07555A. Observes that SGTS Clg OA Dmp HD-07555A opens for approximately 70 seconds.		
*6.	Start SGTS.	Places the selector switch for SGTS Fan OV109A in the START position.		
7.	Monitor the air flowrate.	Monitors SGTS AIR FLOW FR07553A for a flowrate of >3,000 CFM.		

* - Critical Step # - Critical Sequence

PERFORMANCE CHECKLIST

Appl. To/JPM No: NRC 1-#7

Candidate Name:_______________

Step	Action	Standard	Eval	Comments
8.	Check SGTS damper position.	Confirms that the following dampers are in the indicated position: • SGTS Makeup OA Dmp FD07551A2 - MODULATING • SGTS Fan Inlet Dmp HD07552A - OPEN • SGTS A Inlet Dmp HD07553A - OPEN		
9	Refers to OP-173-003, Section 3.3 to vent the drywell	Refers to OP-173-003, Section 3.3		
*10.	Open the Drywell/Wetwell Burp Dampers (HD-17508A & B)	Switches for HD-17508A & B) placed in OPEN		
*11.	Open The Drywell Vent Isolation Valve (HV-15713)	Switch for HV-15713 placed in OPEN		
*12.	Open the Drywell Vent Bypass Outboard Isolation Valve (HV-15711)	Switch for HV-15711 placed in OPEN		
13.	Monitor drywell pressure Evaluator - Inform Candidate that another operator will complete the venting	Monitors pressure on PI-15702 with HSS-15702 selected to CONTN		

* - Critical Step # - Critical Sequence

TASK CONDITIONS:

A. The Unit 1 Drywell pressure has slowly been rising over the last 2 shifts

B. The SGTS is aligned for automatic initiation in accordance with OP-070-001.

C. All prerequisites, Tech Spec and TRM requirements have been met.

INITIATING CUE:

The Unit Supervisor directs you to perform a manual startup of "A" Standby Gas Treatment train and vent the drywell to reduce pressure by 0.5 psig

TASK CONDITIONS:

A. The Unit 1 Drywell pressure has slowly been rising over the last 2 shifts

B. The SGTS is aligned for automatic initiation in accordance with OP-070-001.

C. All prerequisites, Tech Spec and TRM requirements have been met.

INITIATING CUE:

The Unit Supervisor directs you to perform a manual startup of "A" Standby Gas Treatment train and vent the drywell to reduce pressure by 0.5 psig

Appl. To/JPM No: NRC 1-#7 Candidate Name:______________________

QUESTION NO: 1

With the plant operating at power, both trains of Standby Gas Treatment have been declared "Inoperable". What are the restrictions on continued plant operation for these conditions? How does this impact the Secondary Containment?

EXPECTED ANSWER:

- Must restore one SGTS subsystem to Operable in 4 hours or be in Mode 3 in next 12 hours and Mode 4 within 36 hours.
- With both SGTS subsystems Inop, Secondary Containment is Inop. Same actions and completion times as SGTS.

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 261000G112 2.9/4.0

REFERENCES: Unit 1 Tech Spec 3.6.4.1 and 3.6.4.3, Pages 3.6-35 & 42

SY017 L-3, "Standby Gas Treatment System", Rev. 2, LO - 12

Appl. To/JPM No: NRC 1-#7 Candidate Name:______________________

QUESTION NO: 2

A loss of coolant accident with confirmed fuel damage has occurred on Unit 1. All plant systems responded as designed during and after the accident.

What is the expected maximum dose expected to be received at the site boundary?

EXPECTED ANSWER:

25 rem whole body or 300 rem to the thyroid from iodine

ACTUAL ANSWER:

SAT ________ UNSAT ________

K/A NUMBER: 261000A103 3.2/3.8

REFERENCES: 10CFR100.11, "Determination of Exclusion Area, Low Population Zone And Population Zone Center Distance", Rev. 1-1-92 Edition, Section (a)(1) Page 417

SY017 L-3, "Standby Gas Treatment System", Rev. 2, LO - 1, 2 & 3.d

ORIG. SUBMITTAL

Susquehanna S.E.S.
1999 NRC Exam

Nuclear
Department

Simulator Scenarios

T03 1

PP&L–SUSQUEHANNA TRAINING CENTER

SIMULATOR SCENARIO

Scenario Title:	INITIAL LICENSE SIMULATOR EXAM # 1
Scenario Duration:	90 MINUTES
Scenario Number:	99NRC1
Revision/Date:	REV. 1, 3/31/99
Course:	SM100, INITIAL LICENSE EXAM
Operational Activities:	

Prepared By:	Terry W. Logsdon Instructor	3/31/99 Date
Reviewed By:	Nuclear Operations Training Supervisor	Date
Approved By:	Supervising Manager/Shift Supervisor	Date

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SCENARIO SUMMARY

The scenario begins with Unit 1 at 90% power. Unit 2 is 1 hour from synchronizing to the grid. Fuel handling is in progress in Unit 1 Spent Fuel Pool. Instrument Air compressor 'B' is out of service for rebuild. SRV 'R' is leaking. Reactor Recirc 'B' pump is experiencing seal oscillations accompanied by seal stage Hi/Lo flow alarms. SBGT 'B' in service for SO-070-001.

The crew will complete the SBGT surveillance activity on the 'B' train. When the crew shuts down the train the fan inlet damper will fail to close, the SRO will determine the SBGT system is operable in this condition.

HPCI will inadvertently start and inject to the RPV. The crew will attempt to override HPCI injection, a controller malfunction will eventually require isolation of HPCI steam supply isolation valves to terminate injection. HPCI will remain inoperable.

Main turbine oil temperature increases due to a faulty Temperature Control Valve (TCV). A power reduction will be performed to reduce main turbine load. Along with elevated temperatures, main turbine bearing vibration will increase to require a manual reactor scram and main turbine trip. A failure to scram occurs when the Mode Switch is placed to shutdown.

During the response to the ATWS event, ARI and SLC will fail, drifting control rods will be prevented. The crew will lower RPV level with feedwater to the target level control band and override RCIC injection. After control is established in the target band the outboard MSIVs will close. Insufficient high pressure makeup will result in reactor water level decreasing below top of active fuel. The crew will stop and prevent injection except from CRD, initiate Rapid Depressurization, and restore adequate core cooling with Condensate or RHR LPCI injection.

The crew will be unable to manually drive rods due to a failure of Reactor Manual Control System. When RPV level is restored >-161", full rod insertion will be accomplished by isolating and venting the scram air header.

The scenario will terminate when all control rods are inserted, reactor water level is restoring +13 to +54 inches, and direction is given to align Suppression Pool Cooling.

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SCENARIO OBJECTIVES

The SRO candidate will:

1. Ensure Plant Operates IAW the Operating License and Technical Specifications (00.TS.001)
2. Ensure that Required Actions per Technical Specifications/Technical Requirements are met when a LCO/TRO is entered (00.TS.003)
3. Inform other shift members and plant management of changes in plant status, potential plant problems or limitations. (00.AD.131)
4. Implement Scram (00.ON.018)
5. Implement RPV Control (00.EO.026)
6. Implement Level / Power Control (00.EO.031)
7. Implement Primary Containment Control (00.EO.027)
8. Ensure that Required Actions per Technical Specifications are met when a LCO is not met (00.TS.002)
9. Shutdown the reactor when it is determined reactor safety is in jeopardy, or when operating parameters exceed any RPS setpoint and scram does not occur. (00.AD.031)
10. Implement Main Turbine trip. (93.ON.006)
11. Implement appropriate portions of Power Maneuvers (00.GO.010)
12. Implement appropriate portions of Station Communication Practices (00.AD.016)
13. Implement appropriate portions of Operation shift Policies and Work Practices (00.AD.131)

The RO candidate will:

1. Perform operation of RHR in Suppression Pool Cooling with a LPCI signal present (49.OP.012)
2. Perform maximizing CRD flow (55.OP.001)
3. Perform initiation of Standby Liquid Control System (53.OP.003)
4. Perform inserting manual scram with CRD in service (55.OP.006)
5. Perform inhibiting ADS (83.OP.005)
6. Implement Scram (00.ON.018)
7. Implement RPV Control (00.EO.026)
8. Implement Level / Power Control (00.EO.031)
9. Implement Primary Containment Control (00.EO.027)
10. Implement Main Turbine trip. (93.ON.006)
11. Implement appropriate portions of Power Maneuvers (00.GO.010)
12. Perform Monthly Operational Check of SBGT System (70.SO.001)
13. Perform overriding HPCI System (52.OP.009)
14. Perform a 10% power change with Recirc or Rods (00.GO.012)
15. Perform Bypassing MSIV and CIG Interlocks (84.OP.001)
16. Perform overriding RCIC System (50.OP.003)
17. Perform overriding Core Spray System (51.OP.004)
18. Perform overriding RHR System (49.OP.011)
19. Perform initiation of SLC (53.OP.002)
20. Perform Manual operation of ADS (83.OP.001)
21. Perform manual bypass of RWM (31.OP.001)
22. Implement appropriate portions of Station Communication Practices (00.AD.016)
23. Implement appropriate portions of Operation shift Policies and Work Practices (00.AD.131)

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SCENARIO REFERENCES

1. SO-070-001 SGT SURVEILLANCE
   a. SO-070-001 MONTHLY STANDBY GAS TREATMENT
   b. T.S. 3.6.4 STANDBY GAS TREATMENT SYSTEM
   c. NDAP-QA-0321 SECONDARY CONTAINMENT INTEGRITY CONTROL
2. INADVERTENT HPCI INITIATION
   a. OP-152-001 HPCI SYSTEM OPERATION
   b. T.S. 3.5.1 EMERGENCY CORE COOLING SYSTEMS
   c. AR-101-B05 RB AREA PANEL 1C605 HI RADIATION
3. MAIN TURBINE BEARING HIGH TEMPERATURE/HIGH VIBRATION
   a. AR-123-H05 MTLO COOLER DSCH HI TEMP
   b. AR-105-C05 TURB GEN BRG HI TEMP
   c. AR-105-E05 TURB GEN BRG HI VIBR
   d. ON-100-101 REACTOR SCRAM
   e. ON-193-002 MAIN TURBINE TRIP
4. RPV CONTROL
   a. EO-100-102 RPV CONTROL
5. FAILURE TO SCRAM - ATWS
   a. EO-100-113 LEVEL POWER CONTROL
   b. OP-150-001 RCIC SYSTEM OPERATION
   c. OP-155-001 CRD SYSTEM OPERATION
   d. OP-149-001 RHR SYSTEM OPERATION
   e. OP-151-001 CORE SPRAY SYSTEM OPERATION
   f. OP-145-001 FEEDWATER SYSTEM OPERATION
   g. OP-144-001 CONDENSATE SYSTEM OPERATION
6. RAPID DEPRESSURIZATION
   a. EO-100-112 RAPID DEPRESSURIZATION
7. PRIMARY CONTAINMENT CONTROL
   a. EO-100-103 PRIMARY CONTAINMENT CONTROL
   b. OP-116-001 RHRSW SYSTEM OPERATION
   c. OP-149-005 RHR SPC OPERATION
8. OP-AD-001 OPERATIONS SHIFT POLICIES AND WORK PRACTICES

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SCENARIO SPECIAL INSTRUCTIONS

1. Initialize simulator to IC-18, 100% power.
2. Manually start SBGT 'B' system.
3. Lower core flow until APRM power is 90%.
4. Reduce main turbine load set as necessary.
5. Raise DW pressure ≈0.2 psig above existing pressure with nitrogen makeup.
6. Place IA Compressor 'B' control switch to 'OFF' and Pink Tag.
    a. Run the exam initial condition batch file **bat YPB.NRC**
    b. Ensure Main Steam SRV Leaking annunciator is in alarm - AR-110-E01

NOTE: Ensure SBGT 'B' dampers have aligned and operation is stable before continuing.

7. Enter preference file: **restorepref YPP.99NRC1**
    a. Verify environment window

MALFS	**REMFS**	**OVRDS**	**TRG**
6:6	1	1:1	2

    b. Ensure 15 function buttons lit.
8. Silence and reset alarms.
9. Prepare a turnover sheet indicating:
    a. Fuel handling is in progress in Unit 1 fuel pool.
    b. Instrument Air compressor 'B' is out-of-service for rebuild.
    c. SRV 'R' is leaking, tailpipe temp is steady at ≈300°F.
    d. RRP 'B' is experiencing occasional seal oscillations accompanied with seal stage Hi/Lo flow alarms.
    e. SBGT 'B' is in service for SO-070-001, data recording is complete and step 6.3.9 is ready to be performed.
    f. Unit 2 start-up is in progress, approximately 2 hours from synchronizing to the grid.
10. Place simulator in **RUN**.

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SCENARIO EVENT DESCRIPTION FORM

Initial Conditions: Scenario special instructions are complete. Provide the crew with the turnover information. Assign shift positions. Direct the crew to begin the five minute panel walk down.

EVENT	TIME	DESCRIPTION
1		Complete SBGT Surveillance
2		Inadvertent HPCI Start
3		Main Turbine Bearing High Temperature/High Vibration
4		Level/Power Control
5		Outboard MSIVs Close / RPV Level <TAF
6		Rapid Depressurization
7		Reactor shutdown / Primary Containment Control
		Termination Cue

SCENARIO EVENT FORM

Event No: 1
Brief Description: Complete Surveillance SO-070-001 / HD-07552B fails to close

POSITION	TIME	STUDENT ACTIVITIES
SRO		Directs PCO to perform SO-070-001 step 6.3.9
BOPRO		Refers to SO-070-001 step 6.3.9
		Places SGTS fan OV-109B to STOP
		Recognizes fan inlet damper HD-07552B fails to close
		Notifies SRO HD-07552B failed to close
		Refers to AR-130-C10, SGTS OA MU DMP FAIL OPEN
		Dispatch NPO to locally determine damper positions
SRO		Refers to T.S. 3.6.4
		Refers to NDAP-QA-0321 Att. 'A' , SECONDARY CONTAINMENT INTEGRITY
	Note 1	Declares SGTS B is operable and LCO is met.
		Directs PCO to place SGTS fan OV109B to AUTO LEAD
		Identifies fuel handling may continue in this condition
		Contacts maintenance to investigate the failure of HD-07552B damper
BOPRO		Places SGTS fan 0V-109B to Auto Lead

★ Denotes Critical Task

NOTES:	1. Corrective maintenance may lead to LCO not being met, therefore, Required Action A.1 would be entered.

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 1
Brief Description: Complete Surveillance SO-070-001 / HD-07552B fails to close

INSTRUCTOR ACTIVITY:

None

ROLE PLAY:

1. If dispatched as NPO to locally check position of fan inlet damper HD-07552B, wait ≈2 mins. and report the damper is open and there are no mechanical obstructions.

2. If dispatched as maintenance to investigate HD-07552B damper, wait ≈5 mins. and report the damper appear bound open. We will get a work plan approved to perform the necessary repairs. No time estimate is available at this time.

SCENARIO EVENT FORM

Event No: 2
Brief Description: Inadvertent HPCI Initiation and Injection

POSITION	TIME	STUDENT ACTIVITIES
BOPRO		Recognizes and reports HPCI has initiated
		Determines HPCI initiation is not valid by observing RWL indication and Drywell pressure indication
		Refers to OP-152-001 Section 3.9 to override HPCI
SRO		Determines HPCI mis-operation in Auto
		Directs RO to override HPCI injection
		Directs RO to monitor reactor power
BOPRO		Takes action to override HPCI injection
		Notifies SRO HPCI is not responding
RO		Monitors APRM and thermal power change
		Directs NPO to reset 1C605 Rad Monitors
SRO		Directs isolation of HPCI
		Directs RO to monitor MSL and Off-gas rad levels
		Monitors MSL and Off-gas Rad Monitors
BOPRO		Depresses isolation pushbutton and verifies HPCI F003 shuts, turbine trips and injection stops
SRO		Call I&C to investigate HPCI problem
		Refers to T.S. 3.5.1
		Declares HPCI inoperable and declares LCO not met
		Enter RA D.1, verify RCIC is operable immediately and D.2, restore HPCI Operable in 14 days

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 2
Brief Description: **Inadvertent HPCI Initiation and Injection**

INSTRUCTOR ACTIVITY:

1. When actions are complete for SGTS, insert the following to cause HPCI initiation:

   [P-1] IMF HP152004 INADVERTENT HPCI START

2. When HPCI low flow annunciator alarms, insert the following to fail the flow controller auto function:

   [P-2] IMF CN02:FCE411R600 89 0 100 CONTROLLER 1R600 AUTO FAILURE

NOTE: Manual control of 1R600 is failed by a pre-inserted malfunction.

3. If requested to reset ARMs at panel 1C605, insert the following:

   [P-3] MRF RM179004 RESET RESET ARMs AT 1C605

ROLE PLAY:

As I & C dispatched to investigate HPCI system, wait ≈5 mins. and report an intermittent ground exists in the logic. Additional investigation is required, no time estimate for restoration is possible at this time.

SCENARIO EVENT FORM

Event No: 3
Brief Description: Main Turbine Bearing Oil High Temperature / High Bearing Vibration

POSITION	TIME	STUDENT ACTIVITIES
RO		Recognizes and reports alarm "MTLO CLR DSCH HI TEMP"
		Reports TIC-10955 controller output is 100%
		Dispatches NPO locally to TCV-10955 to investigate
		Monitors bearing parameters using PICSY formats
		Refers to AR-123-H05 "MTLO CLR DSCH HI TEMP"
RO		Recognizes and reports alarm "TURB GEN BRG HI TEMP"
		Refers to AR-105-C05 "TURB GEN BRG HI TEMP"
SRO		Directs power reduction to limit bearing temps to <250°F
		Notifies plant management
		Contacts maintenance for assistance
RO		Reports bearing vibration increasing on bearings # 4 & 5, >10 mils but < 11 mils
		Reports alarm "TURB GEN BRG HI VIBR"
		Refers to AR-105-E05 "TURB GEN BRG HI VIBR"
SRO		Directs power reduction to limit bearing vibration
		Enters ON-100-101, Rx Scram, directs performance of scram imminent actions
		Directs manual reactor scram and main turbine trip when bearing vibration exceeds 11 mils
		Notify Chemistry, HP, and RE of power reduction

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 3
Brief Description: Main Turbine Bearing Oil High Temperature / High Bearing Vibration

INSTRUCTOR ACTIVITY:

1. When actions are complete for HPCI, insert the following to cause a Main Turbine Bearing High Oil Temperature:

   [P-4] IMF AV04:TV10955 5 0 25 TV10955 FAILS TO 5% OPEN

NOTE: MTLO cooler discharge high temperature alarms in ≈30 seconds, bearing metal temperature reaches ≈190°F in ≈3 minutes.

Limit bearing metal temps to a maximum of 200°F by modifying the failed position of TV-10955.

2. When bearing metal temperature reaches 190°F, insert the following to cause high vibration in bearings 4 & 5:

   [P-5] bat TUB.NRC1A BRG 4 10.3 MILS & BRG 5 10.1 MILS

3. After power is reduced ≈10 %, insert the following to increase bearing 4 & 5 vibration to > 11 mils:

   [P-6] bat TUB.NRC1B BRG 4 11.2 MILS & BRG 5 11.7 MILS

4. After the Main Turbine is tripped, insert the following to decrease bearing 4 & 5 vibration:

   [P-7] bat TUB.NRC1C DELETES BEARING MALFUNCTIONS

5. If the crew fails to trip the Main Turbine, insert the following to cause the Main Turbine to Trip:

   [P-8] IMF TC193001 TRIP MAIN TURBINE

ROLE PLAY:

1. As NPO dispatched to TV-10955, wait 3 minutes and report the valve appears to be near fully closed and there is some service water flow through the valve by its sound.

2. As NPO dispatched to TV-10955 manual bypass (HV-109110), if manual bypass operation is requested acknowledge the direction for manual control, wait ≈3 mins. and report the valve vibrates shut after it is cracked open. You will try to hold it in position.

SCENARIO EVENT FORM

Event No: 4
Brief Description: ATWS/LEVEL POWER CONTROL

POSITION	TIME	STUDENT ACTIVITIES
RO		Performs power reduction per SRO direction
		Refers to GO-100-012, POWER MANEUVERING
		Plots position on Power / Flow map
		Selects a control rod; monitors for core flux oscillations
		Performs scram imminent actions
		Scrams reactor and trips main turbine per SRO direction
		Recognizes/reports failure to scram
		Arms and depresses the RPS manual scram pushbuttons
BOPRO		Initiates ARI; reports ARI has failed
		Ensures Isolations, ECCS Initiations, and Diesel Generator starts
SRO		Enters EO-100-102, RPV Control and exits to EO-100-113, Level Power Control
		Directs SLC initiated and ADS inhibited
		Directs performance of ES-150-002, SLC Injection with RCIC
		Directs insertion of control rods IAW EO-100-113 Sht. 2, Control Rod Insertion
		Directs venting Scram Air Header
		Directs performance of ES-158-001, De-energizing Scram Pilot Solenoids
		Directs bypassing RSCS and RWM and establishing normal CRD system parameters to manually drive control rods
BOPRO		Initiates SLC and inhibits ADS
		Reports SLC injection has failed

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 4
Brief Description: ATWS/LEVEL POWER CONTROL

INSTRUCTOR ACTIVITY:

When the Mode Switch is placed to shutdown, ensure event trigger #1 actuates to throttle CRD to zero and RMCS fails.

MRF RD155023 0
IMF LC156002

ROLE PLAY:

If requested to perform ES-150-002, SLC Injection with RCIC, acknowledge the direction but take no actions.

SCENARIO EVENT FORM

Event No: 4,5
Brief Description: ATWS/LEVEL POWER CONTROL / OTBD MSIVs CLOSE

POSITION	TIME	STUDENT ACTIVITIES
SRO		Directs RPV water level lowered to <-60" but >-161" with Feedwater
		Directs RPV pressure stabilized below 1087 psig
		Directs overriding RCIC
		Directs bypassing MSIV and CIG interlocks
RO		Lowers and controls RPV water level <-60" but >-161" using feedwater
BOPRO		Overrides RCIC by closing the T &TV
		Bypasses MSIV and CIG interlocks IAW OP-184-001, Main Steam System
		Recognizes/reports OTBD MSIVs have closed
SRO		Directs control of RPV level with RCIC
		Directs monitoring RPV water level
		Directs RPV pressure controlled with SRVs <1087 psig
BOPRO		Attempts restoration of RCIC for injection
		Recognizes/reports RCIC T & TV will not open
		Reports corrected fuel zone level after WR level decreases below -145"
		Reports RPV level is <-161"

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 4,5
Brief Description: ATWS/LEVEL POWER CONTROL / OTBD MSIVs CLOSE

INSTRUCTOR ACTIVITIES:

When RCIC T & TV is overridden closed, insert the following to cause the Trip and Throttle Valve to unlatch and prevent RCIC restart:

[P-9] IMF RC150011	DISABLE RCIC T & TV AFTER CLOSING

When RPV water level is stable between -60" and -110", insert the following to cause the Outboard MSIVs and Drain F019 to close and fail FW controller demand to zero:

[P-10] IMF MS183002	OUTBOARD MSIVs CLOSE
[P-11] IMF MV05:HV141F019	F019 CLOSE
[P-15] bat FWB.99NRC1	FW DEMAND TO MINIMUM

If requested to perform ES-158-001, wait ≈10 minutes and call the control room and state you are ready to pull Div 1 RPS fuses. When permission is granted, insert the following to pull Div. 1 RPS fuses:

[P-12] bat RPB.ES158001A	REMOVE DIV 1 RPS FUSES

ROLE PLAY:

As NPO dispatched to RCIC, wait ≈2 mins. and report the linkage is bent and will not engage the Trip and Throttle Valve.

SCENARIO EVENT FORM

Event No: 6
Brief Description: RAPID DEPRESSURIZATION

POSITION	TIME	STUDENT ACTIVITIES
SRO		Directs all injection stopped and prevented except from CRD and SLC
		Enters EO-100-112, Rapid Depressurization
		Directs Suppression Pool level is verified >5'
		Directs Rapid depressurization by opening all ADS SRVs
BOPRO		Stops and prevents injection except from CRD and SLC
		Verifies Suppression Pool level is >5'
		Rapidly depressurizes the reactor by opening ADS SRVs
SRO		Directs slowly restoring RPV level <-60" but >-161" with CRD, SLC, LPCI or Condensate
		Directs RO to monitor core power as injection begins
RO		Aligns feedwater for start up level control
BOPRO		Slowly restore RPV level <-60" but >-161" as directed
RO		Monitors reactor power during injection

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 6
Brief Description: RAPID DEPRESSURIZATION

INSTRUCTOR ACTIVITY:

None

ROLE PLAY:

As Necessary

SCENARIO EVENT FORM

Event No: 7
Brief Description: REACTOR SHUTDOWN

POSITION	TIME	STUDENT ACTIVITIES
RO		Bypasses RSCS and RWM; attempts to establish normal CRD parameters for manual rod insertion
		Recognizes/reports inability to establish normal CRD system parameters but attempts manual rod insertion
		Recognizes/reports unable to manually drive rods, RMCS is failed
		Inserts control rods IAW EO-100-113 Sheet 2
		Co-ordinates venting Scram Air Header with the NPO
		Verifies control rod insertion as Scram Air Header is vented
		Verifies all control rods are fully inserted
SRO		Directs SLC injection be terminated
		Exits EO-100-113 Sheets 1 and 2; re-enter EO-100-102
		Directs establishing RPV water level +13" to +54"
		Enters EO-100-103, Primary Containment Control
		Directs maximizing Suppression Pool Cooling
RO		Terminates SLC injection
		Establishes RPV water level +13" to +54" with CRD, LPCI or Condensate
BOPRO		Places both loops of RHR in Suppression Pool Cooling IAW OP-149-005, RHR Operation in the Suppression Pool Cooling Mode

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 7
Brief Description: REACTOR SHUTDOWN

INSTRUCTOR ACTIVITY:

1. When the crew has performed Rapid Depressurization and RPV water level is restored to >-161" insert the following to vent the Scram Air Header:

   [P-13] bat RDB.VSAH VENTS SCRAM AIR HEADER

2. If directed to restore the Scram Air Header following venting, wait ≈2 mins. and insert the following:

   [P-14] bat RDB.RSAH RESTORES SCRAM AIR HEADER

ROLE PLAY:

1. As NPO venting the Scram Air Header, inform the crew that you have closed/checked closed HV-147002A/B and uncapped and opened HV-147007. Air has rushed out of the header and has now stopped.

2. As NPO directed to restore the Scram Air Header, wait ≈2 mins. and report you have closed and capped HV-147007 and re-opened HV-147002A, which was the supply valve that was open previously.

TERMINATION CUE:

All control rods are inserted, reactor water level is restoring +13" to +54", and direction is given to align Suppression Pool Cooling.

TA032

PP&L–SUSQUEHANNA TRAINING CENTER

SIMULATOR SCENARIO

Scenario Title:	INITIAL LICENSE SIMULATOR EXAM #2
Scenario Duration:	90 MINUTES
Scenario Number:	99NRC2
Revision/Date:	REV.1, 3/31/99
Course:	SM100, INITIAL LICENSE EXAM
Operational Activities:	

Prepared By:	Terry W. Logsdon Instructor	3/31/99 Date
Reviewed By:	Nuclear Operations Training Supervisor	Date
Approved By:	Supervising Manager/Shift Supervisor	Date

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SCENARIO SUMMARY

The scenario begins with Unit 1 at 65% power. Unit 2 is 1 hour from synchronizing to the grid. Fuel handling is in progress in Unit 1 Spent Fuel Pool. Instrument Air compressor 'B' is out of service for rebuild. SRV 'R' is leaking. Reactor Recirc "B' is experiencing seal oscillations accompanied by seal stage Hi/Lo flow alarms. D/G 'A' is paralleled to bus 1A201 for performance of SO-024-001.

The crew will complete the D/G surveillance and shutdown D/G 'A'. Once the D/G cooldown is started the diesel will trip and be declared inoperable. Direction may be given to substitute D/G 'E' for D/G 'A'.

Reactor recirc pump 'B' lower seal failure will occur. The crew will monitor for changes in leakage into the drywell equipment drain tank.

While the seal failure investigation is continuing the controlling feedwater level channel will drift low resulting in RPV water level increasing. The crew will respond by taking manual control of feedwater injection and transfer control to the backup water level channel. Feedwater control can then be transferred back to automatic.

Reactor Recirc pump 'B' upper seal fails resulting in drywell pressure increase. The crew will reduce power in preparation for removing the pump from service. The crew will evaluate plant conditions and decide to trip Reactor Recirc pump 'B' or perform an orderly shutdown of the pump. Once the pump is stopped the crew will isolate the pump to reduce leakage. When the crew attempts to close the suction valve F023B will fail to close. Drywell pressure will continue to increase, the crew will perform scram imminent action and manually scram the reactor.

Feedwater will initially be available to maintain RPV water level until plant aux load shed trips condensate pumps when drywell pressure exceeds 1.72 psig. HPCI fails during initiation and can not be recovered. RCIC and CRD can be started for injection. D/G 'C' will fail to start and will not be available.

A LOOP occurs that results in loss of division 1 RHR and Core Spray systems. Division 2 RHR and Core Spray systems will be available after D/Gs energize the ESS buses. Containment control will require use of sprays for pressure and temperature control. The leakage rate will increase beyond RCIC and CRD makeup ability and RPV level decreases below TAF. Rapid Depressurization will be required to recover adequate core cooling using low pressure ECCS systems.

Auto opening of RHR injection valve F015B fails and requires operator action to manually open for injection.

The scenario terminates when the reactor is depressurized, reactor water level is restoring +13 to +54 inches, and containment control actions are being addressed.

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SCENARIO OBJECTIVES

The SRO candidate will:

1. Ensure Plant Operates IAW the Operating License and Technical specifications (00.TS.001)
2. Ensure that Required Actions per Technical Specifications/Technical Requirements are met when a LCO/TRO is entered (00.TS.003)
3. Implement Diesel Generator Trip (24.ON.003)
4. Implement RPV Water Level Control System Malfunction (45.ON.006)
5. Implement Appropriate Portions of Single Loop Operation (00.GO.008)
6. Implement Scram (00.ON.018)
7. Implement RPV Control (00.EO.26)
8. Implement Primary Containment Control (00.EO.027)
9. Implement Loss of Offsite Power (04.ON.009)
10. Implement Appropriate Portions of Reactivity Management and Controls Program (00.AD.047)
11. Implement Rapid Depressurization (00.EO.030)
12. Implement RRP Dual Seal Failure (64.ON.005)
13. Implement appropriate portions of Station Communication Practices (00.AD.016)
14. Implement appropriate portions of Operations Shift Policies and Work Practices (00.AD.131)

The RO candidate will:

1. Perform synchronizing D/G to Grid to Restore Normal Power (24.OP.003)
2. Implement Diesel Generator Trip (24.ON.003)
3. Implement RPV Water Level Control System Malfunction (45.ON.006)
4. Implement Appropriate Portions of Single Loop Operation (00.GO.008)
5. Implement Scram (00.ON.018)
6. Implement RPV Control (00.EO.26)
7. Implement Primary Containment Control (00.EO.027)
8. Implement Loss of Offsite Power (04.ON.009)
9. Perform Auto/Manual Startup of RCIC System (50.OP.010)
10. Perform Maximizing CRD (55.OP.001)
11. Implement Appropriate Portions of Reactivity Management and Controls Program (00.AD.047)
12. Perform RHR Response During Auto Initiation of LPCI Mode of Operation (49.OP.009)
13. Perform Core Spray Response During Auto Initiation (51.OP.007)
14. Implement Rapid Depressurization (00.EO.030)
15. Implement RRP Dual Seal Failure (64.ON.005)
16. Perform RHR Operation in Containment Spray (49.OP.005)
17. Perform Manual Operation of ADS (83.OP.001)
18. Perform Manual Operation of SRVs (83.OP.008)
19. Perform Overriding Core Spray Injection (51.OP.004)
20. Perform Overriding RHR Injection (49.OP.011)
21. Perform RHRSW System Startuo Unit 1 Pump to Unit 1 HX (16.OP.002)
22. Implement appropriate portions of Station Communication Practices (00.AD.016)
23. Implement appropriate portions of Operations Shift Policies and Work Practices (00.AD.131)
24. Implement Alarm Responses as applicable (00.AR.005)

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SCENARIO REFERENCES

1. COMPLETE D/G 'A' SURVEILLANCE AND SHUTDOWN D/G A

    A. SO-024-001 D/G MONTHLY OPERABILITY TEST
    B. ON-024-001 DIESEL GENERATOR TRIP
    C. T.S. 3.8.1 AC SOURCES OPERATING

2. RRP 'B' LOWER SEAL FAILURE

    A. AR-102-G05 RRP 'B' SEAL STAGE HI/LO FLOW
    B. ON-164-003 RRP 'B' DUAL SEAL FAILURE

3. FEEDWATER CHANNEL 'A' DRIFT LOW

    A. ON-145-001 RPV LEVEL CONTROL SYSTEM MALFUNCTION
    B. AR-102-B17 RPV WATER LEVEL HI/LO

4. RRP 'B' UPPER SEAL FAILURE

    A. ON-164-003 RRP 'B' DUAL SEAL FAILURE
    B. AR-102-G04 SEAL LEAKAGE HI/LO
    C. T.S. 3.4.4 RCS OPERATIONAL LEAKAGE
    D. GO-100-009 SINGLE RECIRC LOOP OPERATION

5. DRYWELL PRESSURE INCREASE

    A. ON-100-101 SCRAM
    B. EO-100-002 RPV CONTROL

6. LOSS OF OFFSITE POWER

    A. ON-104-001 UNIT 1 RESPONSE TO LOSS OF OFFSITE POWER

7. LARGE LOCA IN CONTAINMENT

    A. EO-100-103 PRIMARY CONTAINMENT CONTROL

8. RAPID DEPRESSURIZATION

    A. EO-100-112 RAPID DEPRESSURIZATION
    B. OP-149-004 RHR CONTAINMENT SPRAY

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SCENARIO SPECIAL INSTRUCTIONS

1. Initialize simulator to IC-32, 69% power.

2. Using Recirc flow, lower power to 65% on APRMs.

3. Raise DW pressure ≈0.2 psig above existing pressure using nitrogen make up.

4. Align Bus 1A201 as follows:

   a. Start ESW pumps A & B
   b. Start D/G 'A' from 0C653
   c. Parallel D/G 'A' to 1A201, increase load to 4MWe

5. Place IA Compressor 'B' control switch to 'OFF' and Pink Tag.

   a. Run the exam initial condition batch file **bat YPB.NRC**

6. Enter preference file: **restorepref YPP.99NRC2**

   a. Verify environment window

MALFS	REMFS	OVRDS	TRG
4:4	1	0:0	2

   b. Ensure 10 function buttons lit.

7. Add the CRC package to the shutdown section.

8. Silence and reset alarms.

9. Prepare a turnover sheet indicating:

   a. Fuel handling is in progress in Unit 1 fuel pool.
   b. Instrument Air compressor 'B' is out-of-service for rebuild.
   c. SRV 'R' is leaking, tailpipe temp is steady at ≈300°F.
   d. RRP 'B' is experiencing occasional seal oscillations accompanied with seal stage Hi/Lo flow alarms.
   e. Power was reduced for repair of RWCU HX endbell, which has been completed.
   f. No instructions for increasing power have been issued.
   g. SO-024-001, D/G Monthly Operability Test is in progress for D/G 'A'. D/G 'A' is paralleled to bus 1A201, the loaded run time has been met. Complete the surveillance starting a step 6.1.16.t (3)
   h. Unit 2 start-up is in progress, approximately 2 hours from synchronizing to the grid.

10. Place simulator in **RUN.**

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SCENARIO EVENT DESCRIPTION FORM

Initial Conditions: Scenario special instructions are complete. Provide the crew with the turnover information. Assign shift positions. Direct the crew to begin the five minute panel walkdown.

EVENT	TIME	DESCRIPTION
1		COMPLETE D/G 'A' SURVEILLANCE
2		RRP 'B' LOWER SEAL FAILURE
3		FEEDWATER CHANNEL 'A' DRIFTS LOW
4		RRP 'B' UPPER SEAL FAILURE / DUAL SEAL FAILURE
5		DRYWELL PRESSURE INCREASE
6		LOSS OF OFFSITE POWER
7		LARGE BREAK LOCA IN DRYWELL
8		RAPID DEPRESSURIZATION

SCENARIO EVENT FORM

Event No: 1
Brief Description: COMPLETE D/G 'A' SURVEILLANCE

POSITION	TIME	STUDENT ACTIVITIES
SRO		Directs completion of D/G 'A' surveillance
BOPRO		Refers to SO-024-001, step 6.1.16.t (3)
		Reduces D/G load to 380-1000 KW
		Performs a 15 minute run prior to shutdown
		Reduces D/G load 300-500 KW
		Trips breaker 1A20104
		Adjust D/G voltage until ≈4.25 KV
		Performs a 5 minute cooldown
		Recognizes/reports D/G 'A' has tripped
		Dispatches NPO to 0C521A to investigate cause of trip
		Refers to ON-024-001, DIESEL GENERATOR TRIP
SRO		Refers to T.S. 3.8.1, AC SOURCES OPERATING
NOTE 1		Declares D/G 'A' inoperable and LCO not met
		Enters RA B.1, perform SR 3.8.1.1 within 1 hour; RA B.3.1, determine other D/Gs are not inoperable; RA B.4, restore D/G to operable in 72 hours
		Refers to NDAP-QA-312, SAFETY FUNCTION DETERMINATION
		Contacts maintenance to investigate D/G 'A'

★ Denotes Critical Task

NOTES:	NOTE 1: May direct substitution of D/G 'E' for D/G 'A'

**INSTRUCTOR ACTIVITIES, ROLE PLAY,
AND INSTRUCTOR'S PERSONAL NOTES**

Event No: 1
Brief Description: COMPLETE D/G 'A' SURVEILLANCE

INSTRUCTOR ACTIVITY:

1. When D/G 'A' breaker 1A20104 is open and voltage is adjusted for 4.25 KV insert the following to trip D/G 'A':

   [P-1] IMF DG024005A D/G 'A' TRIP

ROLE PLAY:

1. As NPO dispatched to investigate the cause of D/G 'A' trip, wait ≈1 minute and report lockout relay 86E-HR for "DIFFERENTIAL TRIP LOCKOUT RELAY' has tripped.

   As NPO dispatched to reset the lockout relay 86E-HR, report the relay will not reset.

2. As Electrical Maintenance or Meter and Relay Test dispatched to D/G 'A' lockout relay problem, wait ≈5 minutes and report the 86E-HR relay appears bad. Estimate 4 hours to set up and bench test a replacement relay.

SCENARIO EVENT FORM

Event No: 2
Brief Description: RRP 'B' LOWER SEAL AILURE

POSITION	TIME	STUDENT ACTIVITIES
RO		Recognizes/reports AR-102-G05, RRP B SEAL STAGE HI/LO FLOW
		Refers to AR-102-G05
		Recognizes upper seal cavity pressure is high
BOPRO		Monitors DEDT level recorder for a change in leakage rate
		Monitors seal cavity temperature for trend on 1C614
SRO		Directs ROs to monitor changes in leakage rate and containment parameters
		Notifies Operations Duty Manager and system engineer about seal condition
		May refer to ON-164-003, RRP DUAL SEAL FAILURE

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 2
Brief Description: RRP 'B' LOWER SEAL AILURE

INSTRUCTOR ACTIVITY:

When actions are complete for D/G 'A', insert the following to cause the RRP 'B' lower seal to fail:

[P-2] IMF RR164003B 5 0 0 RRP 'B' LOWER SEAL FAILURE

ROLE PLAY:

As necessary

SCENARIO EVENT FORM

Event No: 3
Brief Description: FEEDWATER CHANNEL 'A' DRIFT LOW

POSITION	TIME	STUDENT ACTIVITIES
RO		Recognize/reports RX WATER LEVEL HI-LO ALARM
		Reports RPV water level is high
		Refers to AR-101-B17, RX WATER LEVEL HI-LO ALARM
SRO		Directs RO to take manual control of RFPTs
		Directs restoration of RPV level to ≈35"
		Directs implementation of ON-145-001,RPV LEVEL CONTROL SYSTEM MALFUNCTION
		Contacts I & C to investigate level channel 'A' problem
RO		Takes manual control of RFPTs
		Lowers RPV level to +35 "
		Implements ON-145-001, Section 3.7
		Selects FWLC RPV water level channel 'B'
		Nulls master FWL Controller and places controller in Auto
SRO		Refers to T.S. 3.3.2.2, MT High Water Level Trip Instrumentation
		Declares channel 'A' inoperable
		Declares LCO not met
		Enters RS A.1, place channel in trip within 7 days

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 3

Brief Description: FEEDWATER CHANNEL 'A' DRIFT LOW

INSTRUCTOR ACTIVITY:

1. When the crew completes initial monitoring of the RRP 'B' lower seal failure, insert the following to cause FWLC channel 'A' drift low:

   [P-3] IMF TR02:PDTC321N004A 29.5 3 34.5 FWLC CH 'A' OUTPUT DRIFT TO 29.5"

2. When the "RPV Water Level HI" alarms, insert the following to cause FWLC channel 'A' to drift to 26":

   [P-4] IMF TR02:PDTC321N004A 26 3:00 29.5 FWLC CH 'A' OUTPUT DRIFT TO 26"

ROLE PLAY:

As I&C dispatched to investigate FWLC channel 'A' failure, wait ≈3 mins. and report the 'A' channel differential pressure transmitter is failed and must be replaced. Repair time is estimated at 12 hours.

SCENARIO EVENT FORM

Event No: 4
Brief Description: RRP 'B UPPER SEAL FAILURE / DUAL SEAL FAILURE

POSITION	TIME	STUDENT ACTIVITIES
RO		Recognizes/reports RRP 'B' SEAL LEAKAGE HI FLOW
		Refers to AR-102-G04, reports RRP 'B' SEAL LEAKAGE HI FLOW
SRO		Directs implementation of ON-164-003, RRP DUAL SEAL FAILURE
		Directs RO to monitor drywell parameters
		Directs RO to calculate RCS leakage
		Directs the shutdown and isolation of RRP 'B'
		Refers to T.S. 3.4.4, RCS Operational Leakage
		Notifies Reactor Engineering of intent to S/D and isolate RRP 'B'
RO		Refers to ON-164-003, RRP DUAL SEAL FAILURE
		Plots position on Power to Flow Map during the power reduction
		Decrease core flow to ≥55 mlbm/hr
		Inserts control rods below the 70% rod line if a controlled pump shutdown is performed
		Reduces RRP 'B' speed to ≈30%
		Stops RRP 'B'
		Increases RRP 'A' speed to <80% and total core flow >40 mlbm/hr
		Attempts to isolate RRP 'B' by closing suction valve HV-151F023B
NOTE 1		Recognizes/reports suction valve HV-151F023B has dual indication
BOPRO		Reports DW pressure is increasing faster

★ Denotes Critical Task

NOTES:	1. The crew may continue with isolation of RRP 'B' as directed in ON-164-003.

**INSTRUCTOR ACTIVITIES, ROLE PLAY,
AND INSTRUCTOR'S PERSONAL NOTES**

Event No: 4
Brief Description: RRP 'B UPPER SEAL FAILURE / DUAL SEAL FAILURE

INSTRUCTOR ACTIVITY:

1. When the crew has returned FWLC to Auto, insert the following to fail RRP 'B' upper seal:

   [P-5] IMF RR164004B 2 0 0 RRP 'B' UPPER SEAL FAILURE 2 GPM

NOTE: Inserting this malfunction will slowly raise drywell pressure and require a manual reactor scram.

2. When the RO attempts to close the suction valve HV-151F023B, insert the following to increase the seal leakage rate:

   [P-6] MMF RR164004B 50 1:00 2 RRP 'B' UPPER SEAL FAILURE 50 GPM

3. If directed to close RRP 'B' seal purge supply valve HV-1431F008B, insert the following:

   [P-7] MRF RR164041 CLOSE CRD TO RRP 'B'

ROLE PLAY:

As necessary

SCENARIO EVENT FORM

Event No: 5
Brief Description: DRYWELL PRESSURE INCREASE

POSITION	TIME	STUDENT ACTIVITIES
SRO		Enters ON-100-101 and directs scram imminent actions
		Directs manual reactor scram
		Enters EO-100-102, RPV CONTROL
BOPRO		Transfers Aux Buses 11A and 11B to Tie Bus
RO		Starts MTLO pumps
		Manually scrams reactor; verifies all rods full in
		Inserts SRMs and IRMs
		Aligns FW for start up level control
BOPRO		Reports DW pressure >1.72 psig
		Recognizes/reports HPCI turbine trip
		Recognizes/reports D/G 'C' has failed to start; selects isoch and presses start pushbutton
		Verifies ESW cooling to D/Gs
		Initiates RCIC injection to maintain +13" to +54" if feed and condensate trip
SRO		Directs RPV water level control +13" to +54" with RCIC and CRD
		Directs RPV pressure control <1087 psig
		Directs local start of D/G 'C'
		Enter EO-100-103, Primary Containment Control
		Directs cooldown at <100°F/hr
		Contacts Electrical Maintenance to investigate failure of D/G 'C'

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 5
Brief Description: DRYWELL PRESSURE INCREASE

INSTRUCTOR ACTIVITY:

NOTE: After the reactor scram drywell pressure increases more rapidly as leakage rate increases.

1. When the Mode Switch is placed in shutdown, ensure trigger E1 actuates to insert the bottom head drain line leak:

   IMF RR164010 1 30

2. When HPCI F001 opens, ensure trigger E2 actuates to insert a HPCI turbine trip:

   IMF HP152015

3. When requested to attempt a local start of D/G 'C', wait ≈2 mins., transfer D/G 'C' to local using:

   [P-8] IOR QDI43CMC LOCAL D/G 'C' TO LOCAL

ROLE PLAY:

1. As NPO sent to D/G 'C' to attempt a local start, after transferring to local call the control room and report the local start was not successful.

2. As Electrical Maintenance dispatched to D/G 'C', wait ≈5 minutes and report no cause for the failure can be located and we will continue to investigate.

SCENARIO EVENT FORM

Event No: 5,6
Brief Description: DW PRESSURE INCREASE / LOSS OF OFFSITE POWER

POSITION	TIME	STUDENT ACTIVITIES
BOPRO		Dispatches NPO to attempt local start of D/G 'C'
SRO		Directs use of Suppression Chamber Sprays
		Directs termination of Suppression Chamber sprays before suppression chamber pressure drops to 0 psig
BOPRO		Recognizes/reports Loss of Offsite Power
		Reports 4KV Buses 1B and 1D are energized
SRO		Directs a 1C601 walk down for Initiations, Isolations, and D/G starts
		Directs restarting CRD pump 'B'
		Directs maintaining RPV pressure <1087 psig with SRVs
		Contacts PCC and Electrical Maintenance to investigate the loss of power
BOPRO		Starts CRD pump 'B' as directed
		Maintains RPV pressure <1087 psig with SRVs

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 5,6
Brief Description: DW PRESSURE INCREASE / LOSS OF OFFSITE POWER

INSTRUCTOR ACTIVITY:

After drywell pressure exceeds 1.72 psig and the initial plant assessment is complete, insert the following to cause a loss of offsite power:

[P-9] bat DSB.LOOPT21 LOOP

ROLE PLAY:
As PCC contacted for offsite power information, report a breaker failure in the Montour switchyard is responsible for loss of the 230 KV line.

The 230-500 KV tie line has Supervisory Information that indicates a fault on Auto Transformer T-21. Hazleton Dispatch reports sending a crew to the 230 KV switchyard to investigate why the 230 KV breakers 1W and 1T failed to auto re-close.

SCENARIO EVENT FORM

Event No: 7,8
Brief Description: LOCA IN DW / RAPID DEPRESSURIZATION

POSITION	TIME	STUDENT ACTIVITIES
BOPRO		Reports RPV water level is decreasing
		Verifies all LP ECCS pumps start when level drops below -129"
		Transitions to fuel zone level indication when WR level drops below -145"
		Reports corrected fuel zone RPV level is < -161"
SRO		Enters EO-100-112, RAPID DEPRESSURIZATION when RPV level drops below -161"
		Verifies suppression pool level is >5'
		Directs opening 6 ADS SRVs
		Directs Low Pressure ECCS injection to restore RPV level > -161"
BOPRO		Opens 6 ADS SRVs
		Manually opens RHR injection HV-151F015B when RPV pressure is <436 psig
		Restores RPV level above -161" with LP ECCS injection systems
		Transfer to WR level indication when fuel zone indication is >-110"
SRO		Directs throttling injection to restore and maintain RPV level +13" to +54"
		Directs Core Spray injection for RPV level control
		Directs use of Suppression Chamber Sprays
		Directs termination of Suppression Chamber sprays before suppression chamber pressure drops to 0 psig
BOPRO		Implements OP-149-004, RHR CONTAINMENT SPRAY
		Terminates Suppression Chamber sprays before suppression chamber pressure drops to 0 psig
		Limits suppression chamber spray flow to ≈500 gpm

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 7,8
Brief Description: LOCA IN DW / RAPID DEPRESSURIZATION

INSTRUCTOR ACTIVITY:

When the crew completes the assessment of the electric plant lineup, insert the following to increase the leakage from the RPV:

[P-10] MMF RR164010 100 4:00 1 BOTTOM HEAD LEAKAGE RAMP TO 100%

ROLE PLAY:

As necessary

TERMINATION CUE:

The reactor is depressurized, reactor water level is restoring +13" to +54", and containment control actions are being addressed.

TNQ3

PP&L–SUSQUEHANNA TRAINING CENTER

SIMULATOR SCENARIO

Scenario Title:	INITIAL LICENSE SIMULATOR EXAM #3
Scenario Duration:	90 MINUTES
Scenario Number:	99NRC3
Revision/Date:	REV. 1, 3/31/99
Course:	SM100, INITIAL LICENSE EXAM
Operational Activities:	

Prepared By:	Terry W. Logsdon Instructor	3/31/99 Date
Reviewed By:	Nuclear Operations Training Supervisor	Date
Approved By:	Supervising Manager/Shift Supervisor	Date

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SCENARIO SUMMARY

The scenario begins with Unit 1 at 85% power, Unit 2 is 1 hour from synchronizing to the grid. Fuel handling is in progress in Unit 1 Spent Fuel Pool. Instrument Air compressor 'B' is out of service for rebuild. SRV 'R' is leaking. Reactor Recirc 'B' is experiencing seal oscillations accompanied by seal stage Hi/Lo flow alarms.

Work has been completed on RFPT 'B' control signal failure condition, the crew will restore RFPT 'B' to normal alignment.

During the restoration of RFPT 'B', an inop trip of APRM 'E' will occur. This results in a RPS half scram and control rod block signal. The APRM failure will required the SRO to review Technical Specifications and Technical Requirements and determine the LCO and TRO are both met. The SRO will direct the failed APRM be bypassed and the half scram condition reset.

A momentary loss of 1A204 bus requires the crew to recover several systems. Two significant equipment failures occur, a loss of CRD and a loss of chilled water to the RRP motors. When the crew recovers CRD a single control rod will drift partially into the core. The crew will respond by fully inserting the control rod to '00'. When actions are taken to restore cooling to the RRP motors, HV-18792B1 will remain closed. The crew will reduce power to limit motor heat up while action to restore the failed valve continues.

Steam leakage will occur in the Pipe routing Area from HPCI. The Leakage Detection System will alarm for both HPCI and RCIC Pipe Routing Area high temperatures; the crew will enter the Secondary Containment Control procedure. Pipe Routing Area temperatures continue to increase, eventually tripping the Riley Tempmatics and energizing the Pipe Routing Area timers. The crew will make a decision to isolate either HPCI or RCIC and monitor instrumentation for decreasing temperatures in the Pipe Routing Area. When the crew attempts to manually close HPCI steam supply valves F002 and F003, the valves fail to fully close. The Pipe Routing area temperature continues to increase to maximum safe levels, requiring the crew to enter the RPV Control procedure and scram the reactor. Complicating matters will be a brief failure of the Feedwater Master Level Controller to respond properly in Automatic, resulting in RPV level dropping to < -38", causing auto initiation of RCIC and HPCI. When HPCI auto starts, a steam supply line break in the HPCI Equipment Area will result in two areas in Secondary Containment being greater than maximum safe temperatures, requiring Rapid Depressurization.

The scenario terminates when Rapid Depressurization is complete, RPV water level is restoring +13 to +54 inches, and actions are addressed for suppression pool water high temperature in Primary Containment Control.

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SCENARIO OBJECTIVES

The SRO candidate will:

1. Ensure that Required Actions per Technical Specifications / Technical Requirements are met when a LCO/TRO is entered. (00.TS.003)
2. Implement appropriate portions of Power Maneuvers (00.GO.010)
3. Implement appropriate portions of Station Communication Practices (00.AD.016)
4. Implement appropriate portions of Operations Shift Policies and Work Practices (00.AD.131)
5. Implement RPV Control (00.EO.026)
6. Implement Primary Containment Control (00.EO.027)
7. Implement Scram (00.ON.018)
8. Implement Loss of 4KV Bus (00.ON.011)
9. Implement Loss of CRD System Flow (55.ON.014)
10. Implement Loss of RBCW (34.ON.005)
11. Implement Rod Drift (55.ON.013)
12. Implement Primary Break Outside Drywell (00.EO.023)
13. Implement Secondary Containment Control (00.EO.028)

The RO candidate will:

1. Implement appropriate portions of Power Maneuvers (00.GO.010)
2. Implement appropriate portions of Station Communication Practices (00.AD.016)
3. Implement appropriate portions of Operations Shift Policies and Work Practices (00.AD.131)
4. Implement RPV Control (00.EO.026)
5. Implement Primary Containment Control (00.EO.027)
6. Implement Scram (00.ON.018)
7. Implement Alarm Responses as applicable (00.AR.005)
8. Perform RHR in Containment Suppression Chamber Spray (49.OP.005)
9. Perform a 10% power change with Recirc Flow or Rods (00.GO.012)
10. Perform insert a manual scram with CRD in service (55.OP.006)
11. Operate the Manual Scram Pushbuttons (58.ON.003)
12. Implement Loss of 4KV Bus (00.ON.011)
13. Implement Loss of CRD System Flow (55.ON.014)
14. Implement Loss of RBCW (34.ON.005)
15. Implement Rod Drift (55.ON.013)
16. Implement Primary Break Outside Drywell (00.EO.023)
17. Implement Secondary Containment Control (00.EO.028)
18. Perform RFPT Hydraulic Jack Operation (45.OP.016)
19. Perform Manual Operation of ADS (83.OP.001)

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SCENARIO REFERENCES

1. RFPT ' B' SIGNAL FAILURE
   - A. ON-145-001 RPV LEVEL CONTROL MALFUNCTION
   - B. OP-145-001 RFP & RFP LUBE OIL SYSTEM
2. APRM 'E' INOPERABLE
   - A. AR-103-A01 RPS CHAN A1/A2 AUTO SCRAM
   - B. T.S. 3.3.1.1 RPS INSTRUMENTATION
   - C. T.R.3.1.3 CONTROL ROD BLOCK INSTRUMENTATION
   - D. AR-103-A04 NEUTRON MONCHAN A SYSTEM TRIP
   - E. AR-103-A05 APRM CHAN A,C,E UPSCALE/INOP TRIP
3. MOMENTARY LOSS OF BUS 1A204
   - A. ON-104-204 LOSS OF 4KV BUS 1D
   - B. ON-155-007 LOSS OF CRD SYSTEM FLOW
4. LOSS OF RBCW TO RRP 'B'
   - A. AR-102-F04 RECIRC PUMP B MTR WINDING CLG WATER LO FLOW
5. ROD DRIFT
   - A. ON-155-001 CONTROL ROD PROBLEMS
   - B. AR-104-H05 ROD DRIFT
6. SECONDARY CONTAINMENT CONTROL
   - A. AR-108-E05 RCIC LEAK DETECTION HI TEMP
   - B. AR-114-E05 HPCI LEAK DETECTION HI TEMP
   - C. AR-108-F04 RCIC LEAK DET LOGIC A HI TEMP
   - D. AR-108-F05 RCIC LEAK DET LOGIC B HI TEMP
   - E. AR-114-F04 HPCI LEAK DET LOGIC A HI TEMP
   - F. AR-114-F05 HPCI LEAK DET LOGIC B HI TEMP
   - G. AR-114-A02 HPCI STEAM LINE LOGIC A HI DIFF PRESS
   - H. AR-114-A03 HPCI STEAM LINE LOGIC B HI DIFF PRESS
   - I. EO-100-104 SECONDARY CONTAINMENT CONTROL
   - J. ON-100-101 REACTOR SCRAM
7. RPV CONTROL
   - A. EO-100-102 RPV CONTROL
8. RAPID DEPRESSURIZATION
   - A. EO-100-112 RAPID DEPRESSURIZATION
9. PRIMARY CONTAINMENT CONTROL
   - A. EO-100-103 PRIMARY CONTAINMENT CONTROL
   - B. OP-149-005 RHR OPERATION IN THE SUPPRESSION POOL COOLING MODE

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SCENARIO SPECIAL INSTRUCTIONS

1. Initialize the Simulator to **IC-18**, Unit 1 at 100 percent power.

2. Set-up the simulator for the scenario by performing the following:

   A. Reduce Recirc flow until reactor power is 85% on APRMs.
   B. Raise drywell pressure ≈0.2 psig above existing pressure using nitrogen makeup.
   C. Place CRD Pump 'B' in service.
   D. Place IA Compressor 'B' Control Switch to OFF.
   E. Enter batch file **bat YPB.NRC**
   F. Initiate RFPT 'B' control signal failure as follows:
      1) Insert malfunction **IMF FW145004B**
      2) Place RFP 'B' controller in Manual at 65% output
      3) Lower MSC until in control
      4) Place hydraulic jack ON
      5) Mismatch RFP 'B' flow by .5 mlbm of other pumps
      6) Delete malfunction **DMF FW145004B**

3. Enter Preference File: **restorepref YPP.99NRC3**

   A. Check the Environment Window:

MALFS	**REMFS**	**OVRDS**	**TRIGS**
4:3	**1**	**0:0**	**4**

   B. Ensure 7 Function Buttons lit.

4. Silence and reset alarms.

5. Prepare a turnover sheet indicating:

   A. Unit 1 is in MODE 1 at 85% reactor.
   B. Fuel handling is in progress in Unit 1 fuel pool.
   C. Instrument Air compressor 'B' is out-of-service for rebuild.
   D. SRV 'R' is leaking, tailpipe tempature is steady at ≈300°F.
   E. RRP 'B' is experiencing occasional seal oscillations accompanied with seal stage Hi/Lo flow alarms.
   F. RFPT 'B' is controlling on the MSC and the Hydraulic Jack is "ON". A signal failure condition was repaired and RFPT 'B' should be restored to a normal alignment.
   G. Unit 2 start-up is in progress, approximately 2 hours from synchronizing to the grid.

6. Place the Simulator in **RUN**.

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SCENARIO EVENT DESCRIPTION FORM

Initial Conditions: Scenario special instructions are complete. Provide the crew with the turnover information. Assign shift positions. Direct the crew to begin the five minute panel walk down.

EVENT	TIME	DESCRIPTION
1		RECOVER RFPT 'B' SIGNAL FAILURE
2		APRM 'E' INOP TRIP
3		MOMENTARY LOSS OF 4KV BUS 1A204
4		ROD DRIFT
5		LOSS OF RRP MOTOR COOLING
6		STEAM LEAK IN SECONDARY CONTAINMENT
7		RAPID DEPRESSURIZATION
8		TERMINATION CUE

SCENARIO EVENT FORM

Event No: 1
Brief Description: Recover RFPT 'B' signal failure

POSITION	TIME	STUDENT ACTIVITIES
SRO		Directs recovery of RFPT 'B' signal failure
RO		Refers to OP-145-001, Section 3.16
		Resets RFPT 'B' signal failure
		Raise RFPT 'B' speed controller 1R601B to 100% output
		Adjusts the bias thumbwheel to zero
		Selects hydraulic jack to 'OFF'
		Decreases RFPT 'B' speed controller 1R601B until control is established
		Raises MSC to High Speed Stop (HSS)
		Nulls RFPT "B' speed controller 1R601B and places controller in Auto
		Adjusts bias controls to balance RFP flows

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 1
Brief Description: Recover RFPT 'B' signal failure

INSTRUCTOR ACTIVITY:

None

ROLE PLAY:

As necessary

SCENARIO EVENT FORM

Event No: 2

Brief Description: APRM 'E' INOP TRIP

POSITION	TIME	STUDENT ACTIVITIES
RO		Recognizes/reports RPS half scram
		Refers to AR-103-A01, RPS CHAN A1/A2 AUTO SCRAM
		Refers to AR-103-A04, NEUTRON MONCHAN A SYSTEM TRIP
		Refers to AR-103-A05, APRM CHAN A,C,E UPSCALE/INOP TRIP
		Determine APRM 'E' has an Inop Trip
		Dispatches NPO to determine status of APRM 'E'
SRO		Refers to T.S. 3.3.1.1, Table 3.3.1.1-1 function 2.d.
		Determines LCO is met.
		Refers to T.R. 3.1.3, Table 3.1.3-1 function 1.e.
		Determines TRO is met.
		Directs RO to manually bypass APRM 'E'.
		Directs RO to reset RPS half scram
		Directs I & C to investigate APRM 'E'
RO		Bypasses APRM 'E' as directed by SRO
		Resets RPS half scram as directed by SRO

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 2
Brief Description: APRM 'E' INOP TRIP

INSTRUCTOR ACTIVITY:

When RFPT 'B' is restored to Auto control, insert the following to cause a APRM 'E' Inop Trip condition:

[P-1] IMF NM178014E **APRM 'E' INOP TRIP**

ROLE PLAY:

1. As NPO dispatched to check condition of APRM 'E', wait ≈2 minutes and report the Mode switch is in Operate and Inop light is on for APRM 'E'.

2. When contacted as I & C to investigate the failure of APRM 'E', wait ≈3 minutes and report a power supply is failed and estimated repair time is ≈6 hours.

SCENARIO EVENT FORM

Event No: 3
Brief Description: MOMENTARY LOSS OF 4KV BUS 1A204

POSITION	TIME	STUDENT ACTIVITIES
BOPRO		Recognizes/reports power transfer to bus 1A204
		Determines bus 1A204 is energized
		Recognizes/reports D/G 'D' has started
		Verifies ESW cooling to D/G 'D'
		Dispatches NPO to investigate breaker 1A20409
		Dispatches NPO to D/G 'D' to check proper operation
SRO		Directs recovery IAW ON-104-204, LOSS OF 4KV BUS 1D
		Directs restoration of CRD IAW ON-155-007, LOSS OF CRD SYSTEM FLOW
		Directs restoration of RBCW IAW ON-104-204, LOSS OF 4KV BUS 1D
		Contacts EM to investigate problems with 4KV bus 1A204
BOPRO		Restores CRD as directed by SRO
		Restores RBCW as directed by SRO
		Recognizes/reports HV-18792B1 failed to open
		Verifies Instrument Air is operating
		Verifies CIG is operating
		Verifies RBCW is restored to DW coolers
RO		Recognizes/reports rod drift condition when CRD is restored

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 3
Brief Description: MOMENTARY LOSS OF 4KV BUS 1A204

INSTRUCTOR ACTIVITY:

1. When the crew has completed the APRM 'E' failure, insert the momentary loss of power to 4KV bus 1A204 using:

   [P-2] MRF BR061A20409 TRIP TRIP OPEN BKR 1A20409

2. Immediately after the momentary power loss to bus 1D, on 1C681 verify HV-18792B1 is closed, insert a failure to re-open HV-18792B1 using:

   [P-3] IMF AV04:HV18792B1 0 0 0 HV-18792B1 FAILS CLOSED

3. When either CRD pump is restored and flow returned to ≈63 gpm, insert control rod drift of rod 30-47 to position 38 using:

   [P-4] IMF RD1550043047 (NONE 0 10) 10 ROD 30-47 DRIFT TO POSITION 38

ROLE PLAY:

As Electrical Maintenance dispatched to investigate bus 1A204, wait ≈3 mins. and report there is no apparent reason for breaker 1A20409 trip. We will continue to investigate and keep you updated as we trouble shoot.

SCENARIO EVENT FORM

Event No: 4
Brief Description: RESTART CRD/ROD DRIFT

POSITION	TIME	STUDENT ACTIVITIES
RO		Selects "Display Rods Drifting", identifies rod 30-47 as drifting
		Selects rod 30-47
		Identifies rod at position 38
		Determines rod 30-47 should be at position 48
		Refers to ON-155-001, CONTROL ROD PROBLEMS
		Refers to AR-104-H05, ROD DRIFT
SRO		Directs rod 30-47 insertion to position 00
		Notifies Reactor Engineering
RO		Inserts rod 30-47 to position 00

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 4
Brief Description: RESTART CRD/ROD DRIFT

INSTRUCTOR ACTIVITY:

None

ROLE PLAY:

As Reactor Engineering notified about the rod drift condition, reply an investigation is required as to why the rod drifted. While that is happening I will review options to insert symmetrical rods as well as actions to recover the drifted rod 30-47.

SCENARIO EVENT FORM

Event No: 5
Brief Description: LOSS OF RRP MOTOR COOLING

POSITION	TIME	STUDENT ACTIVITIES
SRO		Directs RO to monitor RRP 'B' temperatures on 1C614 recorder
		Contacts maintenance to investigate failure of HV-18792B1
		Directs Recirc pump speeds reduced before motor reaches 204°F
NOTE 1		Directs reactor scram before motor temp reaches 248°F
		Directs power reduction below the 70% rod line
		Notify Chemistry, HP, and RE about power change
BOPRO		Monitors RRP 'B' temperatures
		Refers to AR-102-F02, RRP B MTR WINDING CLG WATER LO FLOW
RO		Reduces Recirc pump speed as directed
		Maintains total core flow >55 mlbm/hr
		Plots position on Power to Flow map
		Selects control rod to monitor core flux oscillations

★ Denotes Critical Task

NOTES:	1. Scram imminent actions may be performed as temp. approaches 248°F.

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 5
Brief Description: LOSS OF RRP MOTOR COOLING

INSTRUCTOR ACTIVITY:

None

ROLE PLAY:

As maintenance sent to investigate HV-18792B1, wait ≈5 mins. and report the solenoid failed and must be replaced. I estimate a minimum of 4 hours to complete the work.

SCENARIO EVENT FORM

Event No: 6
Brief Description: STEAM LEAK IN SECONDARY CONTAINMENT / PRIMARY BREAK OUTSIDE DRYWELL

POSITION	TIME	STUDENT ACTIVITIES
BOPRO		Recognizes/reports AR-108-E05, RCIC LEAK DETECTION HI TEMP and/or AR-114-E05, HPCI LEAK DETECTION HI TEMP.
		Reports Simplex Area 28/29 719' CTMT ACCESS alarm.
		Checks recorders 1R604 and 1R605 and Riley Tempmatic readings at 1C614; reports elevated temperatures in the HPCI/RCIC Pipe Routing Area.
SRO		Enters EO-100-104, SECONDARY CONTAINMENT CONTROL based on Pipe Routing area temperatures.
		Directs starting ESW and all individual Room Coolers.
BOPRO		Responds to AR-108-F04/F05, RCIC LEAK DET LOGIC A/B HI TEMP and AR-114-F04/F05, HPCI LEAK DET LOGIC A/B HI TEMP.
		Reports Pipe Routing Area timers on 1C614 are energized.
		Starts ESW and individual Room Coolers.
		Recognizes/reports HPCI/RCIC auto-isolation and failure of HPCI to isolate.
SRO		Directs manual isolation of HPCI and/or RCIC.
BOPRO		Attempt to manually isolate HPCI, recognizes/reports HPCI F002 valve indicates full open and F003 has lost indication.
		Dispatches NPO to investigate HPCI F003 breaker 1D264081.
NOTE 1		Reports Pipe Routing temperature is approaching 240°F.

★ Denotes Critical Task

NOTES:	NOTE 1: 240°F = MAX SAFE temperature in pipe routing area.

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 6
Brief Description: STEAM LEAK IN SECONDARY CONTAINMENT / PRIMARY BREAK OUTSIDE DRYWELL

INSTRUCTOR ACTIVITY:

When the crew has completed actions for the power reduction with recirc flow, insert the following to cause a steam leak in the Pipe Routing Area:

[P-5] IMF HP152003 1.5 HPCI STEAM SUPPLY LINE LEAK IN PIPE TUNNEL

NOTE: Indications of HPCI leak in Pipe Routing Area occur ≈2.5 minutes after inserting above malfunction.

ROLE PLAY:

1. As NPO dispatched to check trouble alarm at panel 1C275, Rx. Bldg. HVAC, wait ≈3 minutes and report that the alarm received was BDIDs have closed. Indication at 1C275 that nine BDIDs associated with the RHR pipe rooms indicate closed.
2. As NPO dispatched to check breaker 1D264081, wait ≈2 minutes and report that the breaker is closed and appears to be normal.
3. As Electrical Maintenance dispatch to check breaker 1D264081, wait ≈3 minutes and report that it appears as if the breaker has lost control power. You will need to conduct additional troubleshooting to confirm the failure. There is no time estimate for completion of troubleshooting/repairs at this time.

SCENARIO EVENT FORM

Event No: 6
Brief Description: RPV CONTROL/SECONDARY CONTAINMENT CONTROL

POSITION	TIME	STUDENT ACTIVITIES
SRO		Dispatches Electrical Maintenance to investigate breaker 1D264081.
		May direct scram-imminent actions, if time permits.
		Enters EO-100-102, RPV CONTROL after determining that a primary system is discharging into an area and cannot be isolated.
		Directs Mode Switch to SHUTDOWN when Pipe Routing Area reaches Maximum Safe temperature.
		Directs verification of isolations and initiations.
		Directs maintaining RPV level +13" to +54" using available injection sources.
		Directs maintaining RPV pressure <1087 psig with BPVs.
BOPRO		Verifies isolations, initiations and DG starts
		Verifies auto start and proper operation of RCIC.
RO		Maintains RPV pressure with BPVs as directed.
		Reports RPV level dropped < -38", but is recovering with FW.
BOPRO		Verifies auto start of HPCI; recognizes/reports AR-114-A02/A03, HPCI STEAM LINE LOGIC A/B HI DIFF PRESS and reports HPCI has tripped.
SRO		Directs monitoring of Secondary Containment temperatures to assure two areas are not above Maximum Safe Temperatures.

★ Denotes Critical Task

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 6
Brief Description: RPV CONTROL/SECONDARY CONTAINMENT CONTROL

INSTRUCTOR ACTIVITY:

1. When the Mode Switch is placed in Shutdown, verify Event Triggers activate to cause the following:

   a. FWLC Master Controller 1R600 fails low for approximately 20 seconds, then responds as designed in Automatic (resulting in -38" actuations/isolations).

   b. When HPCI starts, a steam line break in the HPCI room is triggered and the pipe routing area leak severity increases, resulting in HPCI Turbine trip, but the F002 and F003 valves are still failed. HPCI Equipment Area temperatures exceed Max Safe, resulting in two areas inside the Secondary Containment exceeding Table 8 requirements for Max Safe temperatures, requiring Rapid Depressurization.

2. When the event trigger activation is verified, insert the following to ramp the HPCI room temperatures above maximum safe values:

   [P-6] bat HPB.99NRC3 RAMP HPCI RM TEMPS ABOVE MAX SAFE

ROLE PLAY:

1. If Security or Health Physics is requested to check for steam release from HPCI blowout panel, wait ≈2 mins. and report there is storm covers have lifted and steam is exiting the vent plenum.

2. If HP is contacted to perform dose calcs for the HPCI release, acknowledge the request. No feedback will be given.

SCENARIO EVENT FORM

Event No: 7

Brief Description: RAPID DEPRESSURIZATION/PRIMARY CONTAINMENT CONTROL

POSITION	TIME	STUDENT ACTIVITIES
SRO		Performs Rapid Depressurization when two area temperatures reach Maximum Safe temperature.
RO		Reports Simplex Area 25/28 645' HPCI PUMP ROOM alarm.
NOTE 1		Reports HPCI Equipment Area temperature is approaching 300°F.
		Reports HPCI area radiation levels are approaching 100 mrem/hr.
SRO		Enters EO-100-112, RAPID DEPRESSURTIZATION due to second area exceeding Maximum Safe temperature.
		Verifies Suppression Pool level >5'.
		Directs Rapid Derpessurization by opening all ADS valves.
		Determines validity of SPING alarm.
		Monitors SPDS for indications of radiation releases.
BOPRO		Rapid Depressurizes the RPV by opening all ADS valves.
		May direct NPO to back-up ADS with Relay Room switches.

NOTES:	NOTE 1: 300°F = MAX SAFE temperature for HPCI EQUIP. AREA.

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 7
Brief Description: RAPID DEPRESSURIZATION/PRIMARY CONTAINMENT CONTROL

INSTRUCTOR ACTIVITY:

If directed to actuate/back-up ADS from Relay Room, wait ≈2 minutes and insert the following:

[P-7] bat ADB.ADSKEYS SIMULATES OPERATING ADS VALVES FROM LRR

ROLE PLAY:

As NPO dispatched to actuate ADS from the Relay Room, wait ≈2 minutes and report that the 6 ADS valves have been keylocked-open for the Lower Relay Room.

SCENARIO EVENT FORM

Event No: 7
Brief Description: RAPID DEPRESSURIZATION/PRIMARY CONTAINMENT CONTROL

POSITION	TIME	STUDENT ACTIVITIES
SRO		Directs restoring RPV level to +13" to +54".
		Re-enters EO-100-104 due to elevated HPCI area radiation levels.
BOPRO		Restores RPV level to +13" to +54" with available systems.
SRO		Enters EO-100-103, PRIMARY CONTAINMENT CONTROL.
		Directs placing both loops of RHR in Suppression Pool Cooling.
BOPRO		Places both loops of RHR in Suppression Pool Cooling IAW OP-149-005, RHR OPERATION IN THE SUPPRESSION POOL COOLING MODE

NOTES:	

**INSTRUCTOR ACTIVITIES, ROLE PLAY,
AND INSTRUCTOR'S PERSONAL NOTES**

Event No: 7
Brief Description: RAPID DEPRESSURIZATION/PRIMARY CONTAINMENT CONTROL

INSTRUCTOR ACTIVITY:

None

ROLE PLAY:

As necessary

TERMINATION CUE:

Rapid Depressurization is complete, RPV water level is restoring +13" to +54", and actions are addressed for suppression pool water temperature high in Primary Containment Control.

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PP&L–SUSQUEHANNA TRAINING CENTER

SUSQUEHANNA STEAM ELECTRIC STATION

PP&L

SIMULATOR SCENARIO

Scenario Title:	INITIAL LICENSE SIMULATOR EXAM #4
Scenario Duration:	90 Minutes
Scenario Number:	99NRC4
Revision/Date:	1, 3/29/99
Course:	SM001, INITIAL LICENSE EXAM
Operational Activities:	

Prepared By:	Terry W. Logsdon Instructor	3/31/99 Date
Reviewed By:	Nuclear Operations Training Supervisor	Date
Approved By:	Supervising Manager/Shift Supervisor	Date

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SCENARIO SUMMARY

The scenario begins with Unit 1 at 100% power, Unit 2 is 1 hour from synchronizing to the grid. Fuel handling is in progress in Unit 1 Spent Fuel Pool. Instrument Air compressor 'B' is out of service for rebuild. SRV 'R' is leaking. Reactor Recirc 'B' is experiencing seal oscillations accompanied by seal stage Hi/Lo flow alarms. Shutdown RHRSW and ESW following completion of RHR loop 'B' in suppression pool cooling.

The crew will shutdown RHRSW and ESW following completion of RHR loop 'B' in suppression pool cooling. When the RHRSW heat exchanger outlet valve is closed it fails in the closed position. The RHRSW loop will remain inoperable, the SRO will declare the LCO not met and enter a 7 day Completion Time for restoring the loop to operable.

A loss of Extraction Steam to 4C heater occurs. The crew will respond by lowering power by 20% using recirculation flow and complete the response by taking the actions stated in the Off Normal procedure. During the follow up actions core flux oscillations occur, the crew will manually scram the reactor. The mode switch to shutdown fails to scram the reactor, however, the manual scram pushbuttons or ARI will insert the control rods.

Following the scram, the Aux Buses 11A and 11B will fail to transfer and a instrument line break occurs inside the drywell. HPCI auto start function is failed but the system can be started using a component by component start up. RPV water level will be maintained with injection from HPCI, RCIC, CRD and SLC. RPV pressure will be controlled by SRV actuation.

When RHR is started a suction leak will develop. The crew will stop the RHR pumps and isolate the suction valve. Suction isolation will fail and suppression pool level will decrease. The crew should continue with actions to control primary containment pressure and temperature. Suppression pool level will stabilize high enough to avoid Rapid Depressurization.

The scenario will terminate when RPV water level is being maintained >TAF, Suppression Chamber and Drywell sprays have been used to control Primary Containment parameters.

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SCENARIO OBJECTIVES

The **SRO Candidate** will:

1. Ensure that required actions per Technical Specifications/Technical Requirements are met when a LCO/TRO is entered (00.TS.003).
2. Implement Loss of Extraction Steam (47.ON.005).
3. Implement appropriate portions of Power Maneuvers (00.GO.010).
4. Direct Reactor Scram on indication of Core Flux Oscillations (78.ON.003).
5. Implement Scram (00.ON.018).
6. Implement RPV Control (00.EO.026).
7. Implement loss of Auxiliary Buses (03.ON.006).
8. Implement Primary Containment Control (00.EO.027).
9. Implement RPV Water Level Anomaly (45.ON.007).
10. Implement Secondary Containment Control (00.EO.028).
11. Implement appropriate portions of Station Communication Practices (00.AD.016).
12. Implement appropriate portions of Operations Shift Policies and Work Practices (00.AD.131).

The **RO Candidate** will:

1. Place RHRSW in standby readiness (16.OP.001).
2. Shutdown ESW system (54.OP.005).
3. Implement Loss of Extraction Steam (47.ON.005).
4. Perform a 10% power change with Rods/Recirc Flow (00.GO.012).
5. Implement appropriate portions of Power Maneuvers (00.GO.010).
6. Implement Core Flux Oscillations (78.ON.003).
7. Insert a Manual Scram with CRD in service (55.OP.006).
8. Operate the Manual Scram Pushbuttons (58.ON.003).
9. Implement Scram (00.ON.018).
10. Implement RPV Control (00.EO.026).
11. Implement loss of Auxiliary Buses (03.ON.006).
12. Implement Primary Containment Control (00.EO.027).
13. Perform a manual start up of HPCI (52.OP.012).
14. Perform maximizing CRD (55.OP.001).
15. Implement RPV Water Level Anomaly (45.ON.007).
16. Place RHR in Containment Suppression Chamber Spray (49.OP.005).
17. Place RHR in Suppression Pool Cooling (49.OP.003).
18. Implement Secondary Containment Control (00.EO.028).
19. Implement Alarm Responses as applicable (00.AR.005).
20. Implement appropriate portions of Station Communication Practices (00.AD.016).
21. Implement appropriate portions of Operations Shift Policies and Work Practices (00.AD.131).

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SCENARIO REFERENCES

1 SHUTDOWN RHRSW AND ESW FOLLOWING COMPLETION OF 'B' RHR IN SUPPRESSION POOL COOLING

a	OP-149-005	RHR OPERATION IN SUPPRESSION POOL COOLING
b	OP-116-001	RHRSW SYSTEM
c	T.S. 3.7.1	RHRSW SYSTEM AND ULTIMATE HEAT SINK
d	AR-150-B01	RHR SERVICE WATER SYSTEM
e	OP-054-001	EMERGENCY SERVICE WATER SYSTEM

2 LOSS OF EXTRACTION STEAM TO 4C FEEDWATER HEATER

a	ON-147-001	LOSS OF FEEDWATER EXTRACTION STEAM
b	GO-100-012	POWER MANEUVERS

3 CORE FLUX OSCILLATIONS

a	ON-178-002	CORE FLUX OSCILLATIONS

4 FAILURE OF MODE SWITCH

a	ON-100-101	SCRAM
b	E0-100-102	RPV CONTROL

5 LOSS OF AUX BUSES 11A AND 11B

a	ON-103-003	13.8KV BUS 11A & 11B LOSS OF BUS LOAD SHEDDING ON UNDERVOLTAGE
b	EO-100-102	RPV CONTROL

6 INSTRUMENT LINE BREAK INSIDE THE DRYWELL

a	EO-100-103	PRIMARY CONTAINMENT CONTROL
b	ON-145-004	RPV WATER LEVEL ANOMALY

7 HPCI AUTO START FAILURE

a	OP-152-001	HIGH PRESSURE COOLANT INJECTION SYSTEM

8 DECREASING SUPPRESSION POOL LEVEL

a	EO-100-104	SECONDARY CONTAINMENT CONTROL
b	AR-109-H8	RHR LOOP A ROOM FLOODED

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SCENARIO SPECIAL INSTRUCTIONS

1 Initialize simulator to IC-18, 100% power.

2 Place IA Compressor 'B' control switch to 'OFF' and Pink Tag.

3 Run the exam initial condition batch file **bat YPB.NRC**

4 Align ESW and RHRSW
   a Place ESW Pump 'A' and 'B' in service
   b Place RHRSW 'B in service
      1) Enable LOCA Trip
      2) Place rad monitor in service; **MRF RM179006 ONLINE**
      3) Adjust RHRSW loop flow 6 - 9 Kgpm

5 Enter preference file: **restorepref YPP.99NRC4**

   a Verify environment window

MALFS	REMFS	OVRDS	TRG
9:9	2	1:1	3

   b Ensure 7 function buttons lit.

6 Silence and reset alarms.

7 Prepare a turnover sheet indicating:

   a Fuel handling is in progress in Unit 1 fuel pool.
   b Instrument Air compressor 'B' is out-of-service for rebuild.
   c SRV 'R' is leaking, tailpipe temp is steady at ≈300°F.
   d RRP 'B' is experiencing occasional seal oscillations accompanied with seal stage Hi/Lo flow alarms.
   e RHR 'B' was in Suppression Pool Cooling; shutdown RHRSW and ESW.
   f Unit 2 start-up is in progress, approximately 1 hours from synchronizing to the grid.

8 Place simulator in **RUN.**

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SCENARIO EVENT DESCRIPTION FORM

Initial Conditions: Initialize the Simulator to **IC-18**. Place the Simulator to RUN. Ensure the Program Buttons are assigned as indicated on the Special Instructions sheet via the appropriate Preference File. Assign Shift positions. Direct the start of the 5 minute panel walk down.

EVENT	TIME	DESCRIPTION
1		Shutdown RHRSW and ESW following completion of RHR 'B' in Supp Pool Cooling
2		Loss of Extraction Steam to 4C Feedwater Heater
3		Core Flux Oscillations
4		Failure of RPS Mode Switch
5		Loss of Auxiliary Buses 11A and 11B
6		Instrument Line Break inside the Drywell
7		HPCI Auto Start Failure
8		Decreasing Suppression Pool Level

SCENARIO EVENT FORM

Event No: 1
Brief Description: Shutdown RHRSW and ESW following completion of Suppression Pool Cooling.

POSITION	TIME	STUDENT ACTIVITIES
SRO		Direct shutdown of 'B' loop RHRSW and ESW.
		Reviews Technical Requirements TRO 3.8.2.1; Required Action A.1 (8 hour TRO) when motor overload bypass is placed in TEST position.
BOP RO		Reviews OP-149-005 and OP-116-001.
		Stops RHRSW Pump.
		Attempts to isolate B RHRSW valves.
		Recognizes/reports BIS alarm indication for RHRSW valve HV-11215B failure.
		Acknowledges BIS alarm AR-150-B01.
		Recognizes/reports valve had stroked full closed when alarm received.
		Directs NPO to check breaker 1B247012.
SRO		Reviews Technical Specification LCO 3.7.1; determines Required Action A.1 (7 day LCO).
		Request assistance from Electrical Maintenance / EWAC.
BOP RO		Reviews OP-054-001.
		Stops running ESW pumps.

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 1
Brief Description: **Shutdown RHRSW and ESW following completion of Suppression Pool Cooling.**

INSTRUCTOR ACTIVITY:

When HV-11215B is full closed, insert the following to prevent valve opening:

[P-1] IOR ZDIHS11215B1 NORM

ROLE PLAY:

As NPO sent to investigate breaker 1B247012, report that it appears the breaker tripped on thermals and will not reset.

As Electrical Maintenance/EWAC: report that the valve motor operator's closed torque switch has failed and the breaker tripped on thermal overload. The limit switch will need to be replaced and the valve checked for damage. You will provide a time estimate as soon as possible.

SCENARIO EVENT FORM

Event No: 2
Brief Description: Loss of Extraction Steam to 4C Heater.

POSITION	TIME	STUDENT ACTIVITIES
BOP RO		Recognize/report Extraction Steam to 4C Heater isolation valve HV-10241C going closed.
		Dispatch NPO to 1C103 to investigate.
SRO		Implement ON-147-001 and GO-100-012.
RO		Reduce reactor power by 20%.
		Plot position on Power/Flow map.
		Adjust load set as necessary.
		Select a control rod; monitor for Core Flux Oscillations.
BOP RO		Monitor Off Gas and MSL radiation monitors.
		Isolate Extraction Steam to 5C heater.
		Isolate Moisture Separator Drains to 4C heater.
SRO		Notify Chemistry, HP, and RE of power reduction.
		Request assistance from I&C.

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 2

Brief Description: Loss of Extraction Steam to 4C Heater.

INSTRUCTOR ACTIVITY:

When Crew has completed actions for RHRSW 15B valve failure, initiate isolation of Extraction Steam to 4C heater:

[P-2] IMF MV05:HV10241C

ROLE PLAY:

1 As NPO dispatched to 1C103: wait ~2 minutes and report no apparent reason for 41C valve closure, feedwater heating system is responding as expected.

2 As I&C investigating extraction steam isolation: wait ~5 minutes and report no obvious reason for isolation has been found; continuing to investigate/troubleshoot.

SCENARIO EVENT FORM

Event No: 3/4/5

Brief Description: Core Flux Oscillations / Failure of Mode Switch / Loss of Aux Buses 11A&B.

POSITION	TIME	STUDENT ACTIVITIES
RO		Recognize/report APRM/LPRM oscillations.
		Monitor severity of power swings.
SRO		Implement ON-178-002.
		Contact Reactor Engineering.
		Direct power reduction to limit oscillations.
RO		Recognize/report severe flux oscillations, LPRM upscale and downscale indications.
		Place Mode Switch to SHUTDOWN.
SRO		Direct reactor scram.
		Implement ON-100-101 and EO-100-102.
RO		Recognize/report failure of Mode Switch.
		Initiate Manual Scram using Manual Scram Push Buttons.
		Insert SRMs and IRMs.
		Report all rods fully inserted.
BOP RO		Initiate ARI (Note 1).
		Recognize/report loss of Aux Buses 11A and 11B.
		Control RPV level with HPCI and/or RCIC.
		Control RPV pressure with SRVs.
SRO		May direct closing MSIVs since condenser is not available.

NOTES: BOP RO may not initiate ARI if RO reports Scram Push Buttons successfully inserted all control rods.

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 3/4/5
Brief Description: Core Flux Oscillations / Failure of Mode Switch / Loss of Aux Buses 11A&B.

INSTRUCTOR ACTIVITY:

1. After 20% power reduction has been performed, and actions to isolate extraction steam to 5C heater are complete per ON-147-001, initiate mild Core Flux Oscillations:

   [P-3] bat NMB.FLUXOSC1

   NOTE: The oscillation batch file may need to be inserted several times while the Crew investigates; Depress P-2 as necessary.

2. When the Crew has noticed the mild oscillations, initiate severe Core Flux Oscillations:

   [P-4] bat NMB.FLUXOSC3

3. When the Manual Scram Push Buttons are depressed, ensure the Trigger E1 actuates to modify Mode Switch position and insert instrument line break malfunction:

   MOR ZDIHSC72A1S01 SHUTDN
   IMF RR180001 100 15:00

ROLE PLAY:

As Electrical Maintenance/EWAC sent to investigate the Aux Buses: wait ~5 minutes and report that it appears there is a failure in the breaker logic for 1A10104 and 1A10204 preventing breaker closure. More time is needed for investigation/troubleshooting.

SCENARIO EVENT FORM

Event No: 6/7
Brief Description: **Instrument Line Break Iside the Drywell / HPCI Auto Start Failure.**

POSITION	TIME	STUDENT ACTIVITIES
RO/BOP RO		Recognize/report loss of Division 1 level and pressure indications.
		Verify RPV level using redundant level indicators.
		Recognize/report Drywell pressure and temperature slowly increasing.
SRO		Implement ON-145-004.
		Implement EO-100-103.
		Re-enter EO-100-102 on High Drywell pressure.
BOP RO		Recognize/report failure of HPCI to auto start (if not already running).
		Start HPCI component by component per OP-152-001.

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 6/7
Brief Description: **Instrument Line Break Iside the Drywell / HPCI Auto Start Failure.**

INSTRUCTOR ACTIVITY:

After HPCI is started and RPV level is recovering, increase severity of Drywell leak:

[P-5] IMF RR164010 15 8:00

ROLE PLAY:

As necessary.

SCENARIO EVENT FORM

Event No: 8
Brief Description: Decreasing Suppression Pool Level.

POSITION	TIME	STUDENT ACTIVITIES
SRO		Direct Suppression Chamber Sprays using 'A' Loop of RHR.
BOP RO		Aligns 'A' Loop of RHRSW.
		Aligns 'A' Loop of RHR for Suppression Chamber Sprays; starts an RHR pump.
		Recognizes/reports RHR LOOP A ROOM FLOODED alarm.
		Verifies Suppression Pool level decreasing.
		Dispatches NPO to investigate room flood.
		Stops RHR pump, closes F004A and F004C.
		Recognizes/reports failure of 4A to close.
SRO		Implements EO-100-104.
		Directs isolation of 'A' Loop RHR.
		Directs start of ESW and Reactor Building room coolers.
		Requests assistance from Maintenance/EWAC to isolate RHR.
		Directs HPCI isolation if level cannot be maintained above 17 feet.
		Directs 'B' Loop RHR placed in Containment Sprays.

NOTES:	If RHR 'A' is started for SPC before sprays, the same actions will occur.

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 8
Brief Description: Decreasing Suppression Pool Level.

INSTRUCTOR ACTIVITY:

1. When 'A' Loop of RHR is placed in service, insert a break on the RHR suction line:

   [P-6] IMF RH149004A 20 4:00

2. When Suppression Pool level reaches 17 feet, verify Trigger E2 actuates to delete the RHR leak:

   DMF RH149004A

ROLE PLAY:

1. As NPO dispatched to verify the RHR room flood alarm: wait ~3 minutes and report that there is at least 4 inches of water on the floor, you have exited the area and closed the water tight door.
2. As Electrical Maintenance/EWAC dispatched to isolate the RHR 4A valve: acknowledge the order and perform no further action.

SCENARIO EVENT FORM

Event No: 8
Brief Description: Decreasing Suppression Pool Level.

POSITION	TIME	STUDENT ACTIVITIES
RO		Starts ESW and Reactor Building room coolers.
		Monitors Primary and Secondary Containment status.
		Dispatches NPO to F004A breaker, 1B216032.
BOP RO		Places 'B' Loop of RHR in Containment Sprays.
		Isolates HPCI as directed, based on Suppression Pool level.
RO		Maximizes CRD as necessary to maintain RPV level.
		Directs bypass of CRD suction filter, as necessary.

NOTES:	

INSTRUCTOR ACTIVITIES, ROLE PLAY, AND INSTRUCTOR'S PERSONAL NOTES

Event No: 8
Brief Description: Decreasing Suppression Pool Level.

INSTRUCTOR ACTIVITY:

1. If directed to bypass the CRD suction filter:

   [P-7] MRF RD155028 100

2. When Suppression Pool level reaches 17 feet, verify Trigger E2 actuates to delete the RHR leak.

   DMF RH149004A

ROLE PLAY:

As NPO dispatched to check breaker 1B216032, wait ~3 minutes and report the breaker tripped and will not reset.

TERMINATION CUE:

When RPV level is being maintained above TAF with available sources and Suppression Chamber/Drywell Sprays have been utilized for containment control, the scenario may be terminated.

ORIG SUBMITTAL

Susquehanna S.E.S.
1999 NRC Exam

Nuclear Department

Written Exam

Final Summary
5/3/99

SUSQUEHANNA NRC WRITTEN EXAM
POST SUBMITTAL CHANGES

NOTE: NORMAL type indicates exam changes made from additional facility validation completed AFTER the initial exam submittal to the NRC.
BOLD type indicates exam changes made based upon the NRC comments per telcon on 04/15/99.
ITALICIZED type indicated exam changes made after Rev. 1 submittal to the NRC on 04/22/99.
UNDERLINED type indicates exam changes made based upon NRC comments made during their prep week 04/26/99.

NOTE: THE REVIEW FORMAT COPY AND CANDIDATE COPY ENCLOSED HAVE HAD THEIR QUESTION ORDER REARRANGED. BOTH NOW HAVE THE "SYSTEM" QUESTIONS FIRST, FOLLOWED BY THE "PWG" QUESTIONS AND THEN THE "EP&E" QUESTIONS. THIS REQUIRED A CHANGE IN THE QUESTION NUMBERS ON THIS LIST OF WRITTEN EXAMINATION CHANGES. THE ORIGINAL QUESTION NUMBER IS LISTED FIRST WITH NEW QUESTION NUMBER IN PARENTHESES.

1 (41). **NRC comment was that b. was not credible. NDAP-QA-0300 states that if the SS leaves the Control Room the Unit 1 US shall assume the Control Room command function. Therefore it is possible that the Candidates may select b. based upon that if the Unit 1 US has the command function, the Unit 2 US may take responsibility for the plant Common Systems. Changed the wording of the distractor to move the reference to the Shift Supervisor.**

2 (42). Added Attachment H graph as reference, corrected section typo in reference.

3 (43). **Clarified conditions in the stem to ensure Candidate knows that the condition requiring actions for Mode 3 and Mode started at 0300 on May 12th. That is, the initial Required Actions could not be taken to restore the system to Operable status within the allowed Completion Time.**

5 (45). Added new 48 hour 4th distractor and removed the 12 hour distractor. Correct answer is now c. after placing distractors in ascending order by time/date.

6 (46). Added assumption that system is operating to stem. Changed question to specifically ask what the operator is "procedurally directed" to do for these conditions.

7 (47). Changed question to specifically ask what the operator is "procedurally required" to do for these conditions.

8 (48). Section 6.2.2 from NDAP-QA-0312 enclosed to explain the Maximum Out Of Service Time calculations for Supporting/Supported ITS systems. No changes made to question at this time.

9 (49). Changed question to SS responsibility for authorizing entry into High Rad Areas without RWP or work plan.

10 (50). Checked math for ALARA numbers. All correct. Changed correct answer to clearly state that two individuals both installed the shielding and then performed the procedure.

11 (51). Changed question to what is required to approve emergency exposures and what can be directed for exposure if not all approvals have been received.

12 (52). Updated reference information due to a new revision to the reference.

Changed question to a direct Unit Supervisor responsibility since Shift Supervision must approve all permits and if a red tag is on a door knob no entry into that area is allowed for any reason even for operation of system not related to the original permit.

13 (53). Changed d. to ensure that it is obvious that the firewatch can "only" be a HP Tech.

NRC comment that this is a Plant Operator level question. SSES procedures specifically require the Unit Supervisor to brief all plant Firewatches on their duties, requirements, etc. Question was not changed.

15 (55). Changed the assumption to ensure Candidate knows that the EAL conditions **have been met but not yet classified**.

16 (56). **Removed reference to SS and ED in stem. Not needed. Changed distractor a. to be more credible by allowing the downgrade following ENR acknowledgement.**

17 (57). **Removed 4th bullet and changed Cognitive Level from "Memory" to "Comprehension". Changed distractor d. to totally incorrect by directing Bypass Valve opening.**

22 (5). Added "and speed raised" to "c." to allow the speed matching by the "A" Pump **speed increase**.

NRC comment was that b. is not credible. With recirc loop flow mismatch not within limits in 2 hours, must declare the lower flow loop as "not in operation" at which time plant is considered to be in "single loop" with 12 additional hours to establish the single loop limits therefore b. was not changed. Changed distractor d. to discuss allowed operation with mismatched loop flows instead of single loop. Tech Spec numbers given in question reference were checked and are correct.

23 (6). Fixed grammatical error, speed "is" limited.

Added plant conditions for analysis for determination of max allowed pump speed.

24 (7). Rewrote distractor "b." and correct answer "d." to cover the logic loss as well as the breaker interlocks.

28 (11). Changed correct answer to Standby Gas Treatment System. CST tank is not required but level instrumentation is. Too close to call.

29 (12). **Changed distractor d. to the opposite Units' CST being available to provide two choices with reference to CST**

30 (13). Fixed typo in stem 52 minutes vs 43 minutes. Removed tank level from stem to avoid confusion with using EOPs to find 200 gallon tank level for when to stop injection.

34 (17). **Changed question to reflect loss of squib continuity and an inoperable squib. Changed order of choices to keep a. as correct answer. Question is now pretty much exactly the same as used on the 02/98 Hope Creek NRC Exam.**

35 (18). Placed "B" RBM in "bypass" so it would not impact the question. Added "channel" to stem

37 (20). Removed extra space in "d.". Changed stem "B" SRM reading to 1.2 cps to move the value off the line on the Signal-To-Noise Ration vs SRM Counts graph in Tech Specs.

39 (22). Fixed grammatical errors in "a." and "d". Added assumption for a dedicated reference leg to avoid having to list several lowering level indicators.

40 (23). Changed distractor order to make correct answer "a." instead of "d." to balance out correct answer distribution.

Added suppression pool and CST levels to require consideration of the potential RCIC auto suction auto swap conditions

41 (24). Added "will" to "b."

42 (25). **Changed distractors to all DC power sources including one additional 125 VDC bus (ADS logic "B") and the Div I and II 250 VDC buses.**

47 (30). Removed extra "SRV" in question stem and specified both vacuum breakers on a single SRV have failed.

48 (31). **Made choices a. and c. both "load reject circuit" and removed the "load reject" term from the stem.**

49 (32). Removed the "will" from "a."

52 (35). Corrected Panel number in stem.

53 (36). Changed Minimum battery cell specific gravity to 1.176 to make it more out of spec. The correct answer was not supported by the old value.

54 (37). **Modified the three incorrect distractors to make more plausible.**

<u>Changed distractor c. to make more plausible by stating that local operator must start engine (true) and locally close output breaker (not true) on a LOOP.</u>

55 (38). Fixed typo and spacing in "a.".

56 (39). **Broke distractors a. and b. 16 hours to Mode 3 requirement into their 4 and 12 hour components.**

60. **Removed the reference to 3.0" HgA from 3rd bullet.**

61. **Changed c. to remove the non-existent high oil temp trip. Now states that the EO bypasses all RCIC trips.**

63. Fixed typo and spacing in stem

64. **Changed a. from a local turbine trip (which might be possible) to tripping all EHC pumps. Even with pumps off, accumulators will keep oil pressure on the valves for a while and then the valves would drift closed.**

67. Capitalized "require" in question stem.

70. Fixed spacing in "a.".

Made the stem read like a leak is occurring into the drywell instead of a leak out of the drywell.

71. Changed last bullet in stem to reflect that not all sections of ES-134-001 can be completed with LOCA present. Changed the "is" to an "are" in bullet #2

Changed d. to more credible choice by stating that the tripped drywell coolers will not be restarted until drywell spraying has been completed.

72. Changed last bullet in stem to "lower" reflecting a steady state to steady state change in level.

Changed K/A from 295013A102 to 295013G447 and updated written exam outline/model.

78. **Changed c. to "damaged fuel". Removed the "any".**

82. Added reactor water level value to the stem to fully meet the requirements for loss of SDC while in Mode 3.

83. Removed the "No actions required......" from the stem and modified the 4 choices to complete sentences.

Changed b. to a third fully inserted control rod adjacent to another rod that has an accumulator alarm in. On choice d., did not change it as it is plausible since this is the time limit to restore charging header pressure to >940 psig with reactor pressure >900 psig with 2 or more inop rods.

87. Fixed typo in "c.".

94. Corrected Unit1/Unit 2 EOP flowchart numbers

98. Fixed stem typo for EOP step number, fixed reference and explanation of answer typos

SUSQUEHANNA

NRC WRITTEN EXAM

REVIEW FORMAT

Specific Unit Supervisor responsibilities

Select the specific plant conditions requiring the Unit 2 Unit Supervisor to assume full responsibility for the plant Common Systems.

- The Unit 1 Unit Supervisor is out of the Unit 1 "At-The-Controls" area.
- The Shift Supervisor has left the Control Room and has delegated the Control Room command function to the Unit 1 Unit Supervisor.
- Unit 1 is in an "off-normal" condition requiring Unit Supervisor and Shift Supervisor attention.
- Unit 1 is shutdown for a scheduled refueling and maintenance outage.

d | S | Memory | Susquehanna | 5/10/99

Generic Knowledge and Abilities | 1 | 1

GENERIC

2.1 Conduct of Operations

2.1.9 Ability to direct personnel activities inside the control room. | 2.5 | 4.0

a. - not procedurally directed, Unit 1 US is still in Control Room b. - Not procedurally directed, Unit 1 US still maintains the Unit 1 and common systems responsibility c. - not procedurally directed d. - correct answer

Conduct Of Operations	NDAP-QA-0300	4.9.1.b	13	9	
Nuclear Department Admin Procedures	AD044			2	

None

New

Question Source Comments:

1 | | 1

[redacted] LCO 3.0.3 actions

Unit 2 has entered LCO 3.0.3 at 1400, May 10, 1999. Preparations for Unit shut down are in progress.

What are the SSES administrative time guidelines for commencing the power reduction?

Power reduction should begin:

[redacted] immediately.

[redacted] not later than 1500.

[redacted] not later than 1700.

[redacted] not later than 1800.

[redacted] c [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Generic Knowledge and Abilities [redacted] 1 [redacted] 1

GENERIC

2.1 Conduct of Operations

2.1.12 Ability to apply technical specifications for a system. 2.9 4.0

[redacted] a. & b. - 3.0.3 requires preparations to begin within 1 hour c. - correct answer per the guidelines d. - 4 hours, too late by guidelines

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Operations Policies And Work Practices	OP-AD-001	6.23.8.b	54	17	
Nuclear Department Admin Procedures	AD044			2	

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 2 [redacted] [redacted] 2

Tech Spec completion times

Given the following information for a Unit 1 Technical Specification System:

- If the Required Action and associated Completion Time for this System is not met the Unit is required to:
  - Be in Mode 3 in 12 hours
    AND
  - Be in Mode 4 in 36 hours
- The Required Actions and Completion Time were not met at 0300 on May 12th
- Unit 1 reached Mode 3 at 0900 May 12th

When is Unit 1 required to be in Mode 4?

- 0300 May 13th
- 1500 May 13th
- 2100 May 13th
- 0300 May 14th

	b		S		Application		Susquehanna		5/10/99

	Generic Knowledge and Abilities	RO Group:	1	SRO Group:	1

GENERIC

2.1 Conduct of Operations

2.1.12 Ability to apply technical specifications for a system. 2.9 4.0

The individual times to reach required Modes start from the initial time when the Required Action and Completion Times are NOT met, they are not additive a. - 24 hours b. - correct answer, 36 hours from RA and TC not met c. - 36 hours from when Mode 3 reached d. - 48 hours from RA and TC not met

Unit 1 Tech Specs		Example 1.3-1	1.3-3	178	

None

Question Source: New   Question Modification Method:

3   3

[redacted] Reactor Mode Switch to "Startup" approval

Prior to placing the Reactor Mode Switch to "Startup/Hot Standby" during a reactor startup, the Shift Supervisor shall notify and obtain approval from the:

- [redacted] Supervisor - Reactor Engineering.
- [redacted] Operations Supervisor - Nuclear.
- [redacted] Manager - Nuclear Operations.
- [redacted] General Manager - Susquehanna.

[redacted] d [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Generic Knowledge and Abilities [redacted] 1 [redacted] 1

GENERIC

2.1 Conduct of Operations

2.1.14 Knowledge of system status criteria which require the notification of plant personnel. 2.5 3.3

[redacted] a. - not procedurally required b. - alternate for notification of MSS into and out of "Refuel" c. - required to be notified for MSS into and out of "Refuel" d. - correct answer.

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Operations Policies And Work Practices	OP-AD-001	6.11.3	25	17	
Nuclear Department Admin Procedures	AD044			2	

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 4 [redacted] [redacted] 4

[redacted]	User Controlled copies limited to 24 hour use

During a Unit 1 evolution, a procedure must be removed from its Controlled Manual. The Operations Department Clerk is not available to provide the User Controlled copy required. The copy of the procedure was made at 1300 on May 11, 1999.

Which of the following is the maximum expiration date and time allowed for this procedure WITHOUT requiring User Controlled tracking from the Document Control Center?

[redacted] 1900, May 11, 1999

[redacted] 0100, May 12, 1999

[redacted] 0700, May 12, 1999

[redacted] 1300, May 12, 1999

[redacted]	d	[redacted]	S	[redacted]	Memory	[redacted]	Susquehanna	[redacted]	5/10/99	

[redacted]	Generic Knowledge and Abilities	[redacted]	1	[redacted]	1

GENERIC

2.1	Conduct of Operations		
2.1.21	Ability to obtain and verify controlled procedure copy.	3.1	3.2

[redacted]	24 hours allowed without DCS tracking a. - end of current shift b. - 12 hours of use c. - end of next shift d. - correct answer

[redacted]					
Operational Procedure Control	OI-AD-002	4.4.4.a.(3)	8	25	
Nuclear Department Admin Procedures	AD044			2	

Material Required for Examination	None

[redacted]	New	Question Modification Method:	

[redacted]	

[redacted]	

[redacted]	5	[redacted]		[redacted]	5

[redacted] Verifying positions of inaccessible valves

Given the following conditions:

- Unit 2 is operating at 85% power
- Operations is performing a check-off list on a system with manually operated valves in the drywell
- These valves do not have Control Room indications

The operator shall verify the position of these drywell valves by:

[redacted] obtaining their positions as noted on the most current Status Control Log.

[redacted] referring to the most recently completed check-off list on the system

[redacted] verifying system parameters (flow, pressure, etc.) are as expected for the current plant conditions.

[redacted] noting the inaccessible valves for verification on the next planned or un-planned drywell entry.

[redacted] c [redacted] S [redacted] Comprehension [redacted] Susquehanna [redacted] 5/10/99

[redacted] Generic Knowledge and Abilities [redacted] 1 [redacted] 1

GENERIC

2.2 Equipment Control

2.2.13 Knowledge of tagging and clearance procedures. 3.6 3.8

[redacted] a. - valves not necessarily under Status Control b. - not procedurally directed, not a "positive" method of determining valve position c. - correct answer d. - does not provide valve position indication now

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Operations Policies And Work Practices	OP-AD-001	6.27.7.a	66	17	
Nuclear Department Admin Procedures	AD044			2	

[redacted] None

[redacted] NRC Exam Bank [redacted] Significantly Modified

[redacted] River Bend NRC Exam (01/97) - changed question from tag removal to checkoff list, one new distractor, different correct answer

[redacted]	[redacted]

[redacted] 6 [redacted] [redacted] 6

[redacted] Tracking Checkoff List Status Changes

With Unit 1 operating at power, a status change via procedure revision has been made to the Reactor Water Cleanup (RWCU) system Checkoff List. The RWCU Reactor Bottom Head Drain Bypass Valve (144F103) component numerical identification has been changed.

Which of the following describes how this status change is tracked until the drywell is accessible allowing a new checkoff list lineup to be performed?

The RWCU checkoff list status change shall be:

[redacted] placed on the list for completion at the next scheduled or unscheduled outage.

[redacted] tracked in the Unit 1 Unit Supervisor Turnover Sheet.

[redacted] documented on the most recently completed checkoff list for the system.

[redacted] tracked in the Unit 1 LCO/TRO log.

[redacted]	b	[redacted]	S	[redacted]	Memory	[redacted]	Susquehanna	[redacted]	5/10/99

[redacted]	Generic Knowledge and Abilities	[redacted]	1	[redacted]	1

GENERIC			
2.2	Equipment Control		
2.2.14	Knowledge of the process for making configuration changes.	2.1	3.0

[redacted] a. - not required for tracking the change b. - correct answer c. - does not ensure the change will be verified when valve is accessible d. - not procedurally directed

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
System Status And Equipment Control	NDAP-QA-0302	6.5.5.a	37	9	
Nuclear Department Admin Procedures	AD044			2	

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 7 [redacted] [redacted] 7

Maximum Out Of Service Time calculation and application

Given the following information:

- Plant Systems "A" and "B" are required to support the operation of System "C"
- The completion times for restoration of these systems to Operable status are:
  - System "A" - 7 days
  - System "B" - 14 days
  - System "C" - 3 days
- System "A" became Inoperable 4 days ago at 0800
- System "B" became Inoperable today at 0800
- System "A" was restored to Operable status today at 1200

Assuming the "Maximum Out Of Service Time" criteria, when must System "B" be restored to Operable status?

At 0800:

- 6 days from today.
- 10 days from today.
- 14 days from today.
- 17 days from today.

a	S	Application	Susquehanna		5/10/99

Generic Knowledge and Abilities		1		1

GENERIC

2.2	Equipment Control		
2.2.23	Ability to track limiting conditions for operations.	2.6	3.8

With "A" inop, MOST for "C" is 10 days, if "B" goes inop and "A" is restored afterwards, the MOST for "C" remains at 10 days, therefore "B" can only be inop 6 days instead of it's full 14 a. - correct answer b. - System "C" MOST c. - System "B" completion time d. - MOST for "C" if "B" had gone inop first.

Control OF LCO's, TRO's And Safety Function Determination Program	NDAP-QA-0312	6.2.2.d	12	1	

None

New

8				8

[redacted] Entry into an HP Controlled Area from the RCA

Which of the following are the MINIMUM requirements for an individual to enter an HP Controlled rea from the Radiologically Controlled Area (RCA)?

[redacted] Review and sign on the HP Controlled Area RWP and receive a specific area briefing from HP.

[redacted] RCA entry meets the requirements for HP Controlled Area entry. No further actions required.

[redacted] Review and sign on the HP Controlled Area RWP and be escorted by HP.

[redacted] Obtain HP approval and perform a local hand and foot survey with a frisker.

[redacted] d [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Generic Knowledge and Abilities [redacted] 1 [redacted] 1

GENERIC

2.3 Radiological Controls

2.3.1 Knowledge of 10 CFR 20 and related facility radiation control requirements. 2.6 3.0

[redacted] HP Controlled Area is a non-contaminated area within the RCA a. - no such RWP, no briefing required b. - required to frisk out of RCA c. - no such RWP, escort not required d. - correct answer

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Radioactive Contamination Control	NDAP-00-0627	6.10.4 & 6.10.7	19 & 20	8	
Radiological Protection	MA062			2	10 & 12

[redacted] None

[redacted] New Question Modification Method:

[redacted]

[redacted]	[redacted]

[redacted] 9 [redacted] [redacted] 9

[redacted] ALARA considerations for exposure

Given the following conditions:

- Unit 1 is making preparations for performing a procedure on a system in a radiation area with a 75 mr/hour dose rate
- The appropriate radiological precautions have been taken
- An HP Briefing has been completed

Using the As Low As Reasonably Achievable (ALARA) guidelines, which of the following is the PREFERRED method for completing this procedure?

[redacted] One individual performing the procedure in the area for 70 minutes.

[redacted] Two individuals performing the procedure in the area for 25 minutes.

[redacted] One individual installing shielding in the area for 30 minutes then performing the procedure for 45 minutes with a reduced dose rate of 7.5 mr/hour.

[redacted] Two individuals installing shielding in the area for 10 minutes then performing the procedure for 25 minutes with a reduced dose rate of 7.5 mr/hour.

[redacted] d [redacted] S [redacted] Application [redacted] Susquehanna [redacted] 5/10/99

[redacted] Generic Knowledge and Abilities [redacted] 1 [redacted] 1

GENERIC

2.3 Radiological Controls

2.3.2 Knowledge of facility ALARA program. 2.5 2.9

[redacted] a. - 87.5 mr total exposure b. - 62.5 mr total exposure c. - 43.125 mr total exposure d. - 31.25 mr total exposure, correct answer

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
ALARA Program And Policy	NDAP-00-1191	4.14.4	14	6	`
Radiological Protection	MA062			2	7

[redacted] None

[redacted] NRC Exam Bank [redacted] Editorially Modified

[redacted] SSES NRC Exam 06/94 - same concept but different numbers utilized in each distractor

[redacted]	[redacted]

[redacted] 10 [redacted] [redacted] 10

[redacted] Exposure extensions during a declared emergency

A Site Area Emergency has been declared on Unit 1.

What is the MAXIMUM Total Effective Dose Equivalent (TEDE) radiation exposure you can be DIRECTED to receive without having to sign on to a Emergency Exposure Request as a volunteer?

[redacted] 2 Rem

[redacted] 4 Rem

[redacted] 5 Rem

[redacted] 25 Rem

[redacted] b [redacted] 5 [redacted] Memory [redacted] 5/10/99

[redacted] Generic Knowledge and Abilities [redacted] 1 [redacted] 1

GENERIC

2.3 Radiological Controls

2.3.4 Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized. 2.5 3.1

[redacted] a. - Current SSES Admin limit with no extensions b. - correct answer, admin limit with extensions, beyond this requires signing an Emergency Exposure Request form volunteering for additional exposure c. - NRC 10CFR20 annual limit d. - Emergency Exposure limit for life saving

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
PP&L Emergency Personnel Dose Assessment And Protective Action Recommendation Guide	EP-AD-000-125	Step 5	7	6	
-Plant Team Management	EP054			2	4

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 11 [redacted] [redacted] 11

Red tagging components in a High Radiation Area

Which of the following describes the method for blocking (installing a red tag) a component in an area with a 6.5 rem/hour dose rate requiring a Health Physics escort for entry? Assume the component is already in the required position and has remote indication.

The red tag:

- shall be installed on the component by Operations Personnel with the Independent Verification requirements waived.
- shall be held by Health Physics and supplied as part of the Radiation Work Permit briefing to all personnel entering the area.
- may be installed on the knob or handle of the door to the area where the component is located.
- is not required if the Operations Lock is accounted for on the permit.

	c		S		Memory		Susquehanna		5/10/99

	Generic Knowledge and Abilities		1		1

GENERIC

2.3	Radiological Controls		
2.3.10	Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.	2.9	3.3

a. - not specifically required by procedure b. - HP not authorized to hold red tags and provide briefings on equipment status c. - correct answer d. - Only true for double locked doors greater than 10 rem/hour.

Standard Blocking Practices	NDAP-QA-0323	6.2.2	14	11	
Energy Control Process For SO Representative, AUS, US And SS	AD032			1	8.d

None

New — Question Modification Method:

	12				12

[redacted] Firewatch tours in High Radiation Areas

Due to Simplex Fire Protection sensor failure, an hourly firewatch is required in a High Radiation Area.

Which of the following describes the restrictions on these firewatch tours?

The Firewatch individual:

[redacted] shall step into the area, make an observation and exit the area.

[redacted] must be escorted by a Health Physics Technician.

[redacted] shall perform a normal walkthrough inspection of the area if total dose expected to be received is less than 10 mrem.

[redacted] must be a Health Physics Technician.

[redacted] a [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Generic Knowledge and Abilities [redacted] 1 [redacted] 1

GENERIC	
2.4	Emergency Procedures and Plan
2.4.25	Knowledge of fire protection procedures. 2.9 3.4

[redacted] a. - correct answer b. - required for continuous firewatches c. - not a procedural requirement d. - can be HP Tech but not required.

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Simplex Problem/Failure Response	OI-AD-03	4.1.5.e	6	5	
Nuclear Department Admin Procedures	AD044			2	

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 13 [redacted] [redacted] 13

10CFR50.54(x) & (Y) criteria

An emergency on Unit 1 has occurred requiring immediate actions be taken that depart from the requirements of Technical Specifications. No actions consistent with Technical Specifications that can provide adequate equivalent protection are immediately apparent.

Which of the following identifies who is required to approve these actions and the specific conditions allowing the actions to be taken as directed in 10CFR50.54(x) & (y)?

- The Emergency Director (Unit 1 Shift Supervisor) approves actions to be taken to protect the health and safety of the personnel outside the SSES site boundary.
- The Emergency Director (General Manager - SSES) approves actions to be taken to protect the health and safety of the personnel outside the SSES site boundary.
- The Emergency Director (Unit 1 Shift Supervisor) approves actions to be taken to protect the health and safety of the personnel inside the SSES site boundary.
- The Emergency Director (General Manager - SSES) approves actions to be taken to protect the health and safety of the personnel inside the SSES site boundary.

a | S | Memory | Susquehanna | 5/10/99

Generic Knowledge and Abilities | 1 | 1

GENERIC

2.4 Emergency Procedures and Plan

2.4.38 Ability to take actions called for in the facility emergency plan, including (if required)supporting or acting as emergency coordinator. 2.2 4.0

a. - correct answer, SRO protecting the public b. - not an SRO c. - SRO protecting the facility personnel, not per 10CFR50.54(x) d. - not an SRO

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	[illegible]
Conditions Of Licenses	10CFR50	54(x) & (Y)	726	1-1-92	
Nuclear Department Admin Procedures	AD044			2	
Operations Policies And Work Practices	OP-AD-001	6.23.6	52	17	

None

NRC Exam Bank | Editorially Modified

Duane Arnold NRC Exam (07/98) - modified to fit SSES specific titles and positions

14 | | 14

[redacted] Time from meeting EAL to classification on a SAE

What is LATEST time that the Emergency Director shall ensure that the State and Local agencies are notified of an emergency once the conditions for an Emergency Action Level (EAL) have been identified? Assume the EAL was met at 0815.

[redacted] 0830

[redacted] 0845

[redacted] 0915

[redacted] 0945

[redacted]	b	[redacted]	S	[redacted]	Memory	[redacted]	Susquehanna	[redacted]	5/10/99

[redacted] Generic Knowledge and Abilities [redacted] 1 [redacted] 1

GENERIC

2.4 Emergency Procedures and Plan

2.4.40 Knowledge of the SRO's responsibilities in emergency plan implementation. 2.3 4.0

[redacted] Assumes the maximum time from EAL met to classification (15 minutes) and max time to make notifications (15 minutes) a. - 15 minutes b. - correct answer, 30 minutes total c. - 1 hour from EAL met, NRC requires one hour notification from classification d. 1 hour from classification

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Emergency Classification	EP-AD-000-200	Timing of Classification 1.3	1	12	
Emergency Plan - Overview	EP001			3	1

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 15 [redacted] [redacted] 15

[redacted] Classifications following momentary exceeding of EAL

Given the following conditions:

- During a transient Unit 2 momentarily met the conditions requiring a Site Area Emergency
- Prior to the actual classification being made, conditions continued to change such that an Alert is now the appropriate classification

What is the Shift Supervisor (SS), in the Emergency Director role, guidance for the classification of this event?

The SS should classify the event as:

[redacted] a Site Area Emergency, make the appropriate notifications and hold the classification at this level until conditions allow entry into the Restoration Phase.

[redacted] an Alert, but should make note of the momentarily Site Area Emergency conditions on the Emergency Notification Report.

[redacted] a Site Area Emergency, make the appropriate notifications and then downgrade the classification to an Alert as soon as possible with management concurrence.

[redacted] an Alert, but should consider upgrading to the Site Area Emergency once all emergency response facilities are activated.

[redacted]	c	[redacted]	S	[redacted]	Memory	[redacted]	Susquehanna	[redacted]	5/10/99

[redacted] Generic Knowledge and Abilities [redacted] 1 [redacted] 1

GENERIC

2.4 Emergency Procedures and Plan

2.4.41 Knowledge of the emergency action level thresholds and classifications. 2.3 4.1

[redacted] Procedurally directed to declare the SAE then immediately downgrade/terminate but actual practice and procedures require upper management concurrence first a. - not directed to stay at the higher classification, unnecessary e-plan response would result b. - directed to declare the SAE to demonstrate the EAL was exceeded c. - correct answer d. - do not wait for facilities to be activated to declare an event, event is declared first

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Emergency Classification	EP012	III.B.4.e	5	3	2

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 16 [redacted] [redacted] 16

[redacted] Operator actions when an expected automatic action did not occur

Given the following conditions:

- Unit 1 is operating at 100% power
- An Electro-Hydraulic Control (EHC) malfunction has resulted in rapidly rising reactor pressure
- Reactor pressure has reached 1100 psig
- There has been NO response from the Reactor Protection System (RPS)

What are the EXPECTED Unit PCO actions for these conditions?

[redacted] Initiate a manual reactor scram and inform the Unit Supervisor of the condition and the action taken.

[redacted] Immediately lower the setpoint of the Maximum Combined Flow Limiter to reduce reactor pressure.

[redacted] Inform the Unit Supervisor of the condition and initiate a manual reactor scram when directed.

[redacted] Do not initiate a manual reactor scram until the RPS failure has been verified by two separate indications.

[redacted] a [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Generic Knowledge and Abilities [redacted] 1 [redacted] 1

GENERIC

2.4 Emergency Procedures and Plan

4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. 4.0 4.0

[redacted] a. - correct action, operator "shall" initiate manual scram if RPS fails to initiate auto scram b. - MCFL lowering will not lower pressure, not an immediate action c. - not appropriate to wait for an order to scram the reactor for these conditions d. - not procedurally directed for these conditions

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Operations Policies And Work Practices	OP-AD-001	6.8.1	20	17	
Nuclear Department Admin Procedures	AD044			2	

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]

[redacted] 17 [redacted] [redacted] 17

"All rods in" times on a Backup Scram Valve initiated scram

Given the following conditions:

- Unit 1 was operating at 100% power
- Following a valid reactor scram signal the Reactor Protection System was unable to de-energize the 185 individual Scram Pilot Valves
- The Backup Scram Valves did function as designed and all control rods fully inserted

Which of the following would be an indication that the Backup Scram Valves actually accomplished the scram?

a. No hydraulic control unit accumulator fault alarms would be received on the full core display.

b. The total elapsed time from the scram signal to all control rods fully inserted would be noticeably longer.

c. The Scram Discharge Volume Vent and Drain Valves would not reposition.

d. The individual control rod scram speeds would be slower.

b | S | Comprehension | Susquehanna | 5/10/99

Plant Systems | RO Group: 1 | SRO Group: 2

201001 Control Rod Drive Hydraulic System

K4. Knowledge of CONTROL ROD DRIVE HYDRAULIC SYSTEM design feature(s) and/or interlocks which provide for the following:

K4.04 Scramming control rods with inoperative SCRAM solenoid valves (back-up SCRAM valves) | 3.6 | 3.6

a. - HCUs still scram the rods, low pressure alarms will be received b. - correct answer, result of venting the entire scram air header through one opening vice 185 c. - SDV Vent and Drains operate as normal when air header depressurizes d. - once rods start moving, no difference in their speeds

Control Rod Drive Hydraulic System	SY017 K-2	IV.B.5.o.4)b)	30	3	11.e

None

Question Source: NRC Exam Bank | Question Modification Method: Editorially Modified

Grand Gulf NRC Exam (07/95) - cleanup stem and distractors, one new distractor

16 | | 18

[redacted]	Indications of failed open Scram Outlet Valve

Given the following CURRENT full core display parameters for control rod 22-35 that had been at Notch "48".

- Full-In: Illuminated
- Full-Out: NOT Illuminated
- Drifting: Illuminated
- Selected: NOT Illuminated
- Accumulator: NOT Illuminated
- Scram Valves: NOT Illuminated

These conditions are the result of:

[redacted] the Scram Inlet Valve (126) opening.

[redacted] the Scram Outlet Valve (127) opening.

[redacted] the Scram Inlet Valve (126) and Scram Outlet Valve (127) both opening.

[redacted] the control rod being driven to Notch "00" using the "Insert Rod" pushbutton.

[redacted]	b	[redacted]	S	[redacted]	Application	Facility:	Susquehanna	[redacted]	5/10/99

[redacted]	Plant Systems	[redacted]	1	SRO Group:	2

201002	Reactor Manual Control System		
K1.	Knowledge of the physical connections and/or cause- effect relationships between REACTOR MANUAL CONTROL SYSTEM and the following:		
K1.01	Control rod drive hydraulic system	3.2	3.2

[redacted]	a. - may also drive rod in but would result in accumulator fault b. - correct answer, provides vent path off top of CRDM operating piston, rod will drift fully in with reactor pressure c. - normal scram water flow path, would get accumulator fault and scram valves light d. - normal insert method would not get drift alarm

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Control Rod Drive Hydraulic System	SY017 K-2	IV.A.5.b.2)k) & Figure 3	11	3	6.u & 11.e

[redacted]	None

[redacted]	New	[redacted]	

[redacted]	

[redacted]	[redacted]

[redacted]	19	[redacted]		SRO Number:	19

[redacted] Uncoupled control rod indications

Given the following conditions:

- Control rod withdrawals for a Unit 2 reactor startup are in progress
- The Unit PCO is withdrawing control rods to Notch "48" using the Continuous Rod Withdrawal and Withdraw Rod pushbuttons
- When control rod 18-19 is withdrawn the following are received
  - Rod Overtravel alarm
  - Rod position indicates "--"

Which of the following is the cause of these indications?

[redacted] The Reactor Manual Control System Rod Motion Timer has malfunctioned resulting in an "overtravel" condition.

[redacted] The PCO provided a withdraw signal to the rod for an excessive period of time after reaching Notch "48".

[redacted] The control rod drive mechanism is at the "overtravel" position but control rod position is currently unknown.

[redacted] The rod has drifted beyond the last even numbered Notch and is still settling back to Notch "48".

[redacted] c [redacted] S [redacted] Application [redacted] Susquehanna [redacted] 5/10/99

[redacted] Plant Systems [redacted] 2 [redacted] 3

201003 Control Rod and Drive Mechanism

A4. Ability to manually operate and/or monitor in the control room:

4.02 CRD mechanism position: Plant-Specific 3.5 3.5

[redacted] a. & b. - a coupled rod withdrawal past "full out" will not give an "overtravel" condition c. - correct answer, indications of uncoupled rod d. - coupled rod may do this but will not pickup "overtravel"

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Reactor Manual Control System	SY017 K-7	IV.B.1.e	4	1	25

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 20 [redacted] [redacted] 20

[redacted] RWM Insert errors

Given the following conditions:

- Control rod withdrawals for a Unit 1 reactor startup are in progress
- The current Rod Worth Minimizer (RWM) group is Group 1
- Group 1 contains 12 control rods that are to be withdrawn from Notch "00" to Notch "48"
- The first 10 rods have been withdrawn to Notch "48" and the remaining 2 rods to Notch "44"
- A control rod in Group 2 has been selected but NOT withdrawn

For these conditions the RWM will display:

[redacted] two withdraw errors and if a third withdraw error is made further rod withdrawals will be blocked except for the three rods with the withdraw errors.

[redacted] two withdraw errors and further rod withdrawals will be blocked except for the rods with the withdraw errors.

[redacted] two insert errors and if a third insert error is made, further rod withdrawals will be blocked except for the three rods with the insert errors.

[redacted] two insert errors and further rod withdrawals will be blocked except for the rods with the insert errors.

[redacted] c [redacted] S [redacted] Application [redacted] Susquehanna [redacted] 5/10/89

[redacted] Plant Systems [redacted] 2 [redacted] 2

201006 Rod Worth Minimizer System (RWM) (Plant Specific)

A2. Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWM); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.05 Out of sequence rod movement; P-Spec(Not-BWR6) 3.1 3.5

[redacted] 2 Group 1 rods at "44" are "insert errors" that will be displayed but will not cause a block until a 3rd insert error is made a. - these two rods are insert errors, no blocks present b. - these two rods are insert errors, no blocks present c. - correct answer d. - takes one additional insert error to produce rod block

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Rod Worth Minimizer	SY017 K-6	Fact Sheets Def 11 & 12	2	1	13 & 15

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]

[redacted] 21 [redacted] [redacted] 21

Mismatched Recirc flow limitations

Given the following conditions:

- Unit 1 was operating at 80% power
- A logic failure has resulted in the "B" Recirculation Pump running back to the #2 Limiter
- Actual #2 Limiter Runback conditions do NOT exist

Which of the following describes the plant limitations required while operating under these conditions?

- If the "B" Recirculation Pump runback cannot be reset in 2 hours it must be tripped within the next 12 hours.
- Single loop operating restrictions and limitations must be in place within 2 hours.
- The "B" Recirculation Pump runback must be reset or the "A" Pump speed reduced to 45% within 2 hours.
- Single loop operation is not permitted and immediate action must be taken to be in Mode 3 within 12 hours.

	c		S		Application		Susquehanna		5/10/99

	Plant Systems		2		2

202001 Recirculation System

A2. Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

.2.08 Recirculation flow mismatch: Plant-Specific | 3.1 | 3.4

a. - not a procedural requirement b. - have 2 hours to declare loop as "not operating" then 12 additional hours to establish single loop limits c. - correct answer, allowed 2 hours with flow mismatch before declaring "low flow" loop as "not operating" d. - partly true but not for these conditions, have 2 hours with mismatched flows

Unit 1 Tech Specs		3.4.1	3.4-1 - 3	178	
Reactor Recirculation System And Motor Generator Set	SY017 L-8			1	46

Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

New

Record Number: 22 | QO Number: | RO Number: 22

[redacted] Recirc Pump Limiter operations

With NO Reactor Feedwater Pumps operating, Recirculation Pump speed are limited to a MAXIMUM of:

[redacted] 20%

[redacted] 30%

[redacted] 40%

[redacted] 45%

[redacted] b [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Plant Systems [redacted] 1 [redacted] 1

202002 Recirculation Flow Control System

A1. Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION FLOW CONTROL SYSTEM controls including:

A1.01 Recirculation pump speed: BWR-2, 3, 4, 5, 6 3.2 3.2

[redacted] With no feed pumps running, feed flow is <20%, therefore Recirc Pumps will be on the #1 Limiter a. - feed flow value that initiates #1 Limiter b. - correct answer c. - scoop tube position for startup d. - #2 limiter

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Reactor Recirculation Control System	SY017 L-9	III.B.2.b.2)	3	1	8 & 20

[redacted] None

[redacted] NRC Exam Bank [redacted] Editorially Modified

[redacted] Hope Creek NRC Exam 02/96 - modified for SSES specific Recirc numbers, changed stem to complete the sentence.

[redacted]	[redacted]

[redacted] 23 [redacted] [redacted] 23

Question Topic: Loss of power to one division of RHR Initiation logic

Given the following conditions:

- Both Units are operating at 100% power
- Unit 2 has Suppression Pool cooling in service on the "A" Residual Heat Removal (RHR) Pump
- A loss of DC power to the Unit 1 RHR Division 1 logic has occurred
- While troubleshooting is in progress a valid loss of coolant accident signal is received on Unit 1

Which of the following describes the expected impact on BOTH Unit's RHR systems?

- The Unit 1 "B" RHR Loop will start and inject normally. The Unit 1 "A" RHR Loop must be manually started and aligned for injection. The Unit 2 "A" RHR Pump will trip.
- All four Unit 1 RHR Pumps will start with injection via both RHR Loops. The Unit 2 "A" RHR Pump must be manually tripped.
- The Unit 1 "B" RHR Loop will start and inject normally. The Unit 1 "A" RHR Loop must be manually started and aligned for injection. The Unit 2 "A" RHR Pump must be manually tripped.
- All four Unit 1 RHR Pumps will start with injection only via the "B" RHR Loop. The Unit 2 "A" RHR Pump will trip.

Answer: d **Exam Level** S **Cognitive Level** Application **Facility:** Susquehanna **Exam Date:** 5/10/99

Tier: Plant Systems **RO Group:** 1 **SRO Group:** 1

203000 RHR/LPCI: Injection Mode (Plant Specific)

2. Knowledge of electrical power supplies to the following:

K2.03 Initiation logic 2.7 2.9

Explanation of Answer: a, b, & c. - Pump start logic cross-divisionalized, Injection valve (F017) logic divisionalized, Unit 2 unaffected d. - correct answer.

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
Residual Heat Removal System Fact Sheets	SY017 C-1	Interlocks 5 & Pump Start Logic 6	3	2	21.b

Material Required for Examination None

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type	Comment

Record Number: 24 **RO Number:** **SRO Number:** 24

Emergency Support Procedure affects on RWCU in Blowdown Mode

Given the following conditions:

- Following a transient, Unit 1 is operating in accordance with EO-100-102, "RPV Control"
- The Pressure Control Leg has directed the use of Reactor Water Cleanup (RWCU) in the Blowdown Mode
- ES-161-001, "RWCU Blowdown Mode Bypassing Interlocks", has been implemented
- Moments after placing RWCU in the Blowdown Mode, a "RWCU System High Leakage" alarm is received and is present for greater than 60 seconds

Select the required operator actions for these conditions assuming RWCU responds as expected.

- Verify automatic closure of the Inboard and Outboard Isolation Valves (F001 and F004).
- Verify automatic closure of the Blowdown Flow Regulator Valve (F033).
- Verify automatic closure of the Inboard and Outboard Isolation Valves (F001 and F004) and the Blowdown Flow Regulator Valve (F033).
- Manually close the Inboard and Outboard Isolation Valves (F001 and F004) and verify automatic closure of the Blowdown Flow Regulator Valve (F033).

a | S | Application | Facility: Susquehanna | 5/10/99

Plant Systems | RO Group: 2 | SRO Group: 2

204000 | Reactor Water Cleanup System

3. | Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including:

A3.03 | Response to system isolations | 3.6 | 3.6

ES-161-001 bypasses only the Filter Demin Inlet high temp isolation (F004 closure), all other isolations remain in effect a. correct answer, normal RWCU isolation on leak b. - only closes on low upstream or high downstream pressure, cannot be assumed for these conditions c. - F033 no auto close for these conditions d. - F001 and 4 auto close, F033 does not

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
Reactor Water Cleanup System	SY017 L-1	VI.D.1	25	1	
Reactor Water Cleanup System	SY017 L-1	Fact Sheets Ops 8.d	5	1	16 & 17

Material Required for Examination: None

Question Source: New | Question Modification Method:

Record Number: 25 | RO Number: | SRO Number: 25

[illegible] LPCI Injection Valve operation following SDC isolation

Given the following conditions:

- Unit 2 is in Mode 4 with Shutdown Cooling in service on the "B" Residual Heat Removal (RHR) loop
- A large leak has developed just upstream of the Shutdown Cooling Suction Outboard Isolation Valve (F008)
- Reactor water level rapidly reaches the Low Pressure Coolant Injection (LPCI) initiation setpoint
- All expected actions occur
- Core Spray is NOT available

Which of the following describes the expected affect on the leak and reactor water level for these conditions?

a. The leak will be stopped and reactor water level will stabilize but not recover unless operator action is taken to inject.

b. The leak will NOT be stopped. Operator action is required to isolate the leak and inject with RHR to recover level.

c. The leak will be stopped and reactor water level will rise due to the "B" Loop of RHR injecting in the LPCI mode.

d. The leak will NOT be stopped. Operator action is required to isolate the leak allowing automatic LPCI injection to recover level.

Answer:	a
Tier/Level:	S
Cognitive Level:	Application
Facility:	Susquehanna
Exam Date:	5/10/99
Tier:	Plant Systems
RO Group:	2
SRO Group:	2

205000	Shutdown Cooling System (RHR Shutdown Cooling Mode)
K3.	Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM/MODE will have on following:
K3.02	Reactor water level: Plant-Specific — 3.2 — 3.3

Explanation of Answer: a. - correct answer, SDC Inboard Suction Valve (F009) auto isolates at +13", but the LPCI Injection Valves 15A & B are overridden closed until reset by the operator, must also realign previously running RHR Pump suction path b. - SDC Inboard Valve (F009) auto closure at +13" will stop leak c. - "B" Loop Injection valve (F015B) will be overridden closed until operator action taken to reset it d. - SDC Inboard Valve (F009) auto closure at +13" will stop leak, operator action required to inject as above

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
Residual Heat Removal System	SY017 C-1	Fact Sheets SDC Interlocks	4	2	8, 9.c & 9.g

Material Required for Examination: None

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

26				26

HPCI Ramp Generator failure will operating in automatic

Given the following conditions:

- The Unit 1 High Pressure Coolant Injection (HPCI) system is running in the CST to CST mode
- The Flow Controller is in "Automatic" set for 3500 gpm
- System flowrate is 3500 gpm
- The Extra PCO reports that HPCI turbine speed is lowering

Which of the following would cause this response?

- A relay failure has just transferred the Flow Controller from "Automatic" to "Manual"
- The HPCI Test Line To CST Isolation Valve (F011) has just auto closed.
- The HPCI ramp generator output just failed to its "low" limit.
- The HPCI Minimum Flow To Suppression Pool Valve (F012) has just opened.

c | S | Application | Susquehanna | 5/10/99

Tier: Plant Systems | RO Group: 1 | SRO Group: 1

206000 | High Pressure Coolant Injection System

K5. | Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM:

.5.05 | Turbine speed control: BWR-2, 3, 4 | 3.3 | 3.3

a. - in "Manual" turbine speed should remain unchanged b. - F011 closing will be sensed as less flow therefore turbine should speed up to raise flow c. - correct answer, the turbine control valve is positioned based upon the lowest of the two signals it receives, flow controller output versus ramp generator output, therefore speed lowers d. - Opening F012 will reduce the flow the flow transmitter is seeing, therefore turbine should speed up to restore flow

High Pressure Coolant Injection System	SY017 C-6	Fact Sheet Operation 1.j. & Figure 18	3	2	7.d & 9

None

New

27				27

HPCI support equipment vs operable

Which of the following High Pressure Coolant Injection (HPCI) "support" systems/components, if Inoperable, would NOT affect the Operability of HPCI?

- The Condensate Storage Tank
- The Auxiliary Oil Pump
- The Suppression Pool
- The Minimum Flow To Suppression Pool Valve (F012)

a		S		Comprehension		Susquehanna		5/10/99	
Plant Systems					1		1		

206000 High Pressure Coolant Injection System

K6. Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI):

K6.09 Condensate storage and transfer system: BWR-2, 3, 4 — 3.5 — 3.5

a. - correct answer, CST not considered by TS for HPCI operability b. - HPCI won't start without Aux Oil Pump c. - the TS HPCI suction source of water d. - affects HPCI components and system flowrates

Unit 1 Tech Specs		3.5.1	3.5-1	178	
High Pressure Coolant Injection System	SY017 C-6			2	11.f & 20

Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

New

28 — 28

Core Spray operability while lined up to the CST

Which of the following conditions MUST be met when the "A" Core Spray loop suction is lined up to the Condensate Storage Tank (CST)? Assume the Unit CSTs are NOT cross-connected.

- The reactor vessel head must be removed and the core defueled.
- The "A" Core Spray loop must be declared Inoperable.
- The Unit Condensate Storage Tank level must be greater than 49%.
- The "B" Core Spray Loop must remain Operable.

c | S | Application | Susquehanna | 5/10/99

Plant Systems | RO Group: 1 | SRO Group: 1

209001 | Low Pressure Core Spray System

K1. | Knowledge of the physical connections and/or cause- effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following:

K1.01 | Condensate storage tank: Plant-Specific | 3.1 | 3.1

a. - allowed do lineup to CST in Modes 4 & 5 b. - not required, TS specifically allows lineup to CST c. - correct answer d. - not required

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
Unit 1Tech Spec		3.5.2	3.5-10	178	
Core Spray	SY017 C-2			2	12

Material Required for Examination: Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

Question Source: New | Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 29 | RO Number: | SRO Number: 29

Question Topic: Cold Shutdown Boron injected criteria

Given the following conditions:

- Unit 2 has experienced a failure-to-scram (ATWS)
- The Standby Liquid Control (SLC) system was initiated and injected for 43 minutes before both SLC Pumps failed
- Reactor power is in the source range
- SLC Storage Tank level is 950 gallons

How does this failure affect the planned reactor cooldown and depressurization?

a. Boron concentration is sufficient to allow a complete cooldown under any plant conditions.

b. Cooldown can be accomplished if completed before Xenon decays out of the core.

c. Boron concentration is sufficient to allow a complete cooldown with a maximum of 8 control rods not fully inserted.

d. Reactor Engineering must make the determination if current boron concentration will allow a complete cooldown.

Answer: a | Exam Level: S | Cognitive Level: Application | Facility: Susquehanna | Exam Date: 5/10/99

Tier: Plant Systems | RO Group: 1 | SRO Group: 1

211000 | Standby Liquid Control System

A2. | Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.01 | Pump trip | 3.5 | 3.8

Explanation of Answer: a. - correct answer, 2 pumps at TS minimum of 41.2 gpm for 52 minutes is 4284 gallons, greater than CSBW of 4191 gallons b. - CSBW will account for Xe decay as well c. - CSBW will handle any number of rods out d. - not required

Reference Title	Facility Reference Number	Section	Page Numbers	Revision	LO
Standby Liquid Control System	SY017 C-3	IV.B & Table 1	6 & 21	2	11.b

Material Required for Examination: None

Question Source: New | Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 30 | RO Number: | SRO Number: 30

SLC "Subsystem" criteria and operability

Given the following conditions:

- Unit 1 is operating at 100% power
- The "B" Standby Liquid Control (SLC) Pump was declared "Inoperable" 4 days ago
- The "Loss Of Continuity To Squib Valves" alarm has just been received
- Investigation reveals broken leads to the "A" SLC Squib Valve primers
- The "B" SLC Squib Valve primer continuity status has not changed

Select the required actions for these conditions.

- Restore one subsystem to Operable status in 8 hours or be in Mode 3 within the next 12 hours.
- Continue in the 7 day Required Action for one Inoperable subsystem, no further actions are required.
- Enter a 7 day Required Action for the "A" SLC Subsystem, continue in the 7 day Required Action for the "B" SLC Subsystem.
- Extend the current 7 day Required Action for one Inoperable subsystem not to exceed 10 days from the initial failure to meet the LCO.

b | S | Application | Susquehanna | 5/10/99

Plant Systems | 1 | 1

211000 Standby Liquid Control System

K3. Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on following:

3.01 Ability to shutdown the reactor in certain conditions 4.3 4.4

With one SLC pump out then taking out one of the two parallel Squib valves still only have one SLC "Subsystem" out of service a. - only one subsystem inop, if both out, these are the actions required b. - correct answer c. - only one subsystem inop, even with both inop, these actions would not be correct d. - does not meet the criteria to extend to 10 days total

Unit 1 Tech Specs		3.1.7	3.1-20	178	
Unit 1 Tech Spec Bases		B 3.1.7	B 3.1-40	0	
Standby Liquid Control System	SY017 C-3			2	4, 14.c & d

Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

Question Source: New

Question Modification Method:

Record Number: 31 RO Number: SRO Number: 31

[redacted] RPS vs Backup Scram Valve relationship

Given the following conditions:

- Unit 2 is operating at 60% power
- A valid reactor scram signal occurs on high drywell pressure

Which of the following failures would PREVENT the Backup Scram Valves from venting the scram air header?

[redacted] The solenoid on the upstream Backup Scram Valve (110B) does not de-energize.

[redacted] The Alternate Rod Injection Scram Air Header Block Valves (SV 14799 & 147100) did not close on the scram.

[redacted] Only one Reactor Protection System Trip System de-energized on the scram signal.

[redacted] The check valve (111) bypassing the downstream Backup Scram Valve (110A) does not open.

[redacted] c [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Plant Systems [redacted] 1 [redacted] 1

212000 Reactor Protection System

A2. Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.19 Partial system activation (half-SCRAM) 3.8 3.9

[redacted] a. - Backup Scram Valve solenoids energize on a scram signal b. - ARI auto initiation does not occur on high drywell pressure, no impact on scram air header, not required to close to allow Backup Scram Valves to vent the scram air header c. - correct answer, both RPS trip systems must trip to energize both Backup Scram Valves to vent the header d. - check valve value would not affect other Backup Scram Valve's ability to vent scram air header

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Reactor Protection System	SY017 L-5	II.B.5 & Figure 3	4	1	14.b & 22.g

[redacted] None

Question Source: Previous 2 NRC Exams Question Modification Method: Editorially Modified

[redacted] SSES NRC Exam 09/96 - changed stem to bullet format, used specific valve numbers in distractors

[redacted]	[redacted]

[redacted] 32 [redacted] [redacted] 32

[illegible]: Actions for Inoperable RPS EPAs

Given the following conditions:

- Unit 1 is operating at 100% power
- The "B" Reactor Protection System (RPS) MG Set is out of service for extended maintenance
- One of the "A" RPS MG Set Electrical Protection Assembly (EPA) output breaker undervoltage relays has been determined to be Inoperable

Select the required actions.

a. Insert a half scram and remove the "A" RPS MG Set from service within 1 hour.

b. Restore the EPA to Operable status within 72 hours or be in Mode 3 in 12 hours and Mode 4 in 36 hours

c. Transfer the "A" RPS Bus to the alternate power supply within 72 hours.

d. Restore the EPA to Operable status within 1 hour or be in Mode 3 in 12 hours and Mode 4 in 36 hours

[illegible]: b | [illegible]: S | [illegible]: Application | Facility: Susquehanna | [illegible]: 5/10/99

[illegible]: Plant Systems | RO Group: 1 | SRO Group: 1

212000 Reactor Protection System

2.1 Conduct of Operations

2.1.12 Ability to apply technical specifications for a system. 2.9 4.0

[illegible]: a. - 1 hour actions are for both power supplies with Inop EPA b. - correct answer c. - correct action if alternate power available, "B" RPS is on alternate power d. - 1 hour actions for both power supplies with Inop EPA

[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Unit 1 Tech Specs		3.3.8.2	3.3-75	178	
Reactor Protection System	SY017 L-5			1	10 & 13.b

[illegible]: Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

Question Source: New | Question Modification Method:

[illegible]:

[illegible]	Comment

[illegible]: 33 | [illegible]: | SRO Number: 33

Question Topic: TIP Panel indications

The following are the current indications on Valve Control Monitor Panel for Channel 1 of the Traversing Incore Probe (TIP) System (see attached figure) :

- Ball Valve "Closed" lights - both illuminated
- Ball Valve "Open" lights - both extinguished
- Shear Valve Monitor Lights - both extinguished
- Squib Monitor lights - both illuminated

Which of the following describes the status of TIP Channel 1's Shear Valves and primary containment integrity?

a. The TIP Shear Valves are operable and primary containment integrity is met.

b. The TIP Shear Valves are inoperable and primary containment integrity is not met.

c. The TIP Shear Valves are inoperable and primary containment integrity is met.

d. The TIP Shear Valves are operable and primary containment integrity is not met.

Answer: a | Exam Level: S | Cognitive Level: Memory | Facility: Susquehanna | ExamDate: 5/10/99

Tier: Plant Systems | RO Group: 3 | SRO Group: 3

215001 Traversing In-Core Probe

K4. Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following:

K4.01 Primary containment isolation: Mark-I&II(Not-BWR1) | 3.4 | 3.5

Explanation of Answer: Shear Valve Monitor out indicates a valve has not actuated therefore shear valve should be operable, ball valve closed assures primary containment integrity is met a. - correct answer b. - Squib Monitor on indicates good squib, ball valves being closed meets primary containment integrity c. - Squib monitor on indicates good squib d. ball valves being closed meets primary containment integrity

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
Traversing Incore Probe System	SY017 I-5	IV.B.3	14 & 15	1	13 & 14

Material Required for Examination: Valve Control Monitor Panel, Figure 14 SY017 I-5

Question Source: NRC Exam Bank | Question Modification Method: Concept Used

Question Source Comments: Hope Creek NRC Exam 02/98 - rewrite stem to reverse the question to have operable squibs as well as primary containment integrity

Comment Type	Comment

Bank Number: 34 | RO Number: | SRO Number: 34

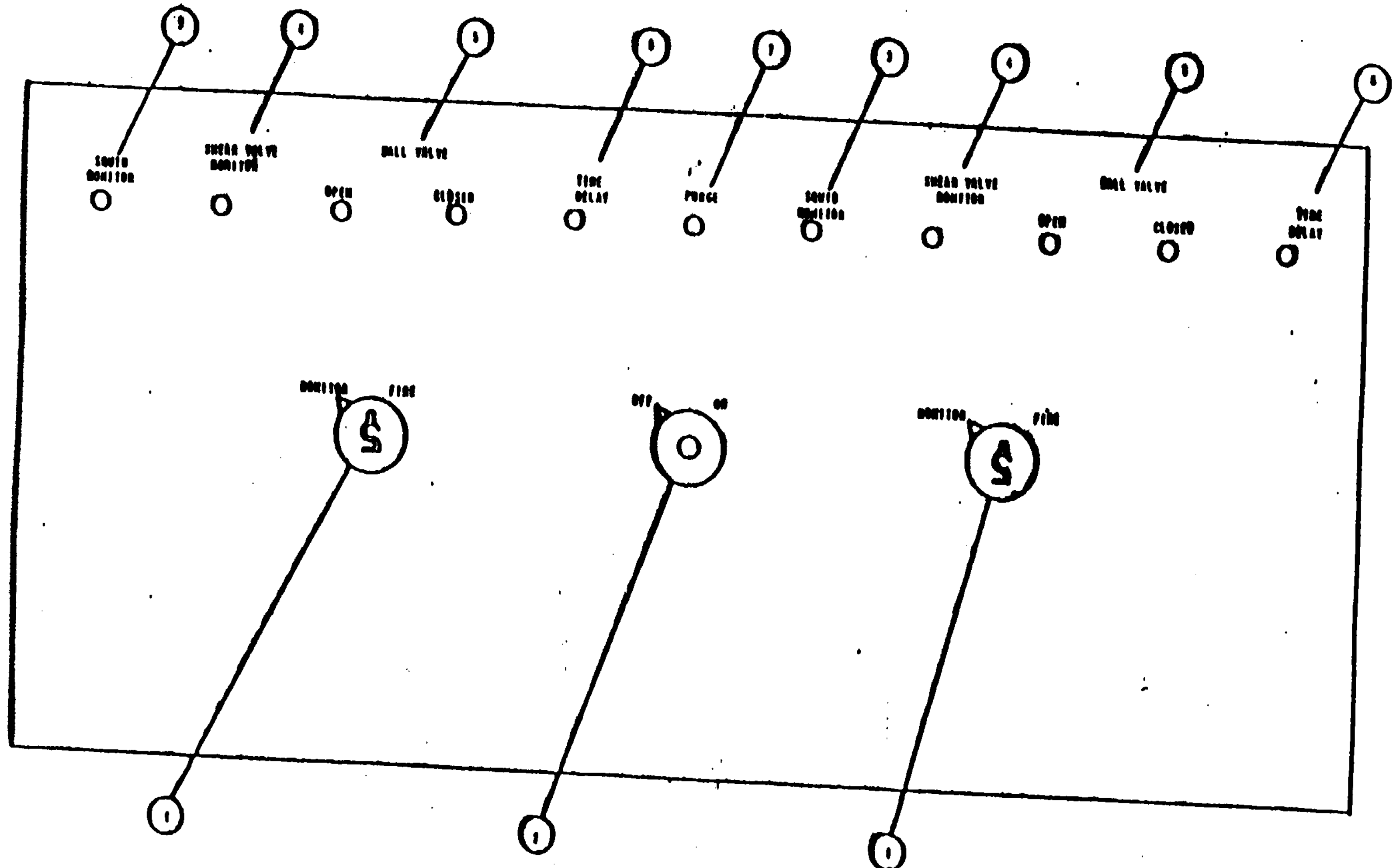


FIGURE 14
DUAL VALVE CONTROL MONITOR

Question Topic: RBM Gain Change Circuit failure

Given the following conditions:

- Unit 2 is at 35% with power ascension in progress
- The "A" Rod Block Monitor (RBM) Gain Change Circuit malfunctions and does NOT provide any LPRM input signal gain adjustments

How does this malfunction affect the continuing reactor startup?

- The RBM channel would default to its low trip setpoint and generate a rod withdrawal block.
- Local power around a withdrawing control rod may reach a higher level before any automatic protective actions occur.
- The RBM channel would transfer to the alternate Reference APRM allowing continued rod withdrawals.
- The local power rise during a control rod withdrawal can only be controlled by the RBM Backup Trip Unit.

Answer:	b	Exam Level:	S	Cognitive Level:	Application	Facility:	Susquehanna	ExamDate:	5/10/99

Tier:	Plant Systems	RO Group:	2	SRO Group:	2

215002	Rod Block Monitor System		
K6.	Knowledge of the effect that a loss or malfunction of the following will have on the ROD BLOCK MONITOR SYSTEM:		
K6.05	LPRM detectors: BWR-3, 4, 5	2.8	3.1

Explanation of Answer: Failure of the gain change circuit could mean that local power would be much lower than core average thus a rod withdrawal could result in a much higher/rapid local power increase before the RBM trips stop the rod a. - no auto default to low trip setpoint b. - correct answer c. - transfer only done via APRM joystick d. - backup trip unit is highest setpoint, other three trip unit should initiate rod block before this one

Reference Title	Facility Reference Number	Section	Page Numbers	Revision	LO
Rod Block Monitor	SY017 K-5	IV.B.3.	5-7	0	6.b

Material Required for Examination: None

Question Source: New **Question Modification Method:**

Question Source Comments:

Comment Type	Comment

Bank Number: 35 **RO Number:** **SRO Number:** 35

Question Topic	IRM Downscale rod blocks

Given the following conditions on Unit 1:

- A reactor startup is in progress with the Reactor Mode Switch in "Startup/Hot Standby"
- All Intermediate Range Monitor (IRM) channels are reading 3/125 on Range 2
- All Average Power Range Monitor (APRM) channels are reading "downscale"
- Both Rod Block Monitor (RBM) channels are reading "downscale"
- The Rod Select Clear pushbutton is illuminated
- All systems are operating as designed

Control rod withdrawals are being prevented by:

a. an RBM rod block.

b. an APRM rod block.

c. a "No Rod Selected" rod block.

d. an IRM rod block.

Answer	d	Exam Level	S	Cognitive Level	Application	Facility:	Susquehanna	Exam Date:	5/10/99
Tier:	Plant Systems	RO Group:	1	SRO Group:	2				

215003	Intermediate Range Monitor (IRM) System		
A4.	Ability to manually operate and/or monitor in the control room:		
A4.07	Verification of proper functioning/ operability	3.6	3.6

Explanation of Answer: a. - bypassed until power is >30% b. - rod block active with RMS in "Run" c. - rod block active with RMS in "Refuel" d. - correct answer, rod block occurs with RMS in "Startup/Hot Standby"

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
Intermediate Range Monitor	SY017 I-2	IV.C.1	19	1	9

Material Required for Examination: None

Question Source: New | Question Modification Method:

Question Source Comments:

Comment Type	Comment

Question Number: 36 | RO Number: | SRO Number: 36

[illegible] SRM operability during fuel load

Given the following conditions:

- Unit 1 is ready to load fuel following a complete core offload (no fuel assemblies are in the core)
- Source Range Monitor (SRM) channels "A" and "C" are Inoperable
- SRM detectors "B" and "D" are fully inserted with count rates of 1 cps and 2 cps respectively
- The Signal-To-Noise Ratio for SRM "B" and "D" is 3:1 for both channels

Which of the following describes what is required to commence core reload under these conditions?

a. One of the Operable SRMs must be in the quadrant where core alterations are being performed and the other in an adjacent quadrant. Minimum count rate is not required to be met until 4 fuel assemblies have been installed adjacent to the SRM.

b. SRM "A" or "C" must be returned to Operability to allow core alterations to begin. Minimum count rate is not required to be met until 4 fuel assemblies have been installed adjacent to the SRM.

c. One of the Operable SRMs must be in the quadrant where core alterations are being performed and the other in an adjacent quadrant. A neutron source must be installed adjacent to the SRMs to achieve a minimum 3 cps.

d. SRM "A" or "C" must be returned to Operability and a neutron source must be installed adjacent to the SRMs to achieve a minimum 3 cps to allow core alterations to begin.

Answer: a | [illegible]: S | [illegible]: Application | Facility: Susquehanna | [illegible]: 5/10/99

[illegible]: Plant Systems | RO Group: 1 | SRO Group: 1

215004 | Source Range Monitor (SRM) System

2.2 | Equipment Control

2.2.26 | Knowledge of refueling administrative requirements. | 2.5 | 3.7

Explanation of Answer: a. - correct answer per Tech Specs b. - not required to have 3 SRM channels operable c. - can be less than 3 cps until 4 fuel assemblies installed adjacent to SRM d. - not required to have 3 SRMs Operable or to have 3 cps

Reference Title	[illegible]	Section	[illegible]	[illegible]	[illegible]
Unit 1 Tech Specs		3.3.1.2	3.3-10 - 3.3-15	176	
Source Range Monitor	SY017 I-1			2	2 & 3

Material Required for Examination: Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

Question Source: New | Question Modification Method:

[illegible]:

Comment Type	Comment

Record Number: 37 | RO Number: | SRO Number: 37

[illegible] APRM Gain Adjustment requirements

Given the following conditions:

- Unit 1 is operating at 100% power
- Average Power Range Monitor (APRM) Channel "C" has been bypassed with the joystick for maintenance

A Gain Adjustment for APRM "C" will be required:

- prior to taking it out of "Bypass".
- if it differs by more than 2% from the average of the remaining 5 APRM channels.
- prior to exceeding a gain adjustment factor (AGAF) of 1.00.
- if its gain adjustment factor (AGAF) is less than 0.98.

[illegible] d [illegible] S [illegible] Memory Facility: Susquehanna [illegible] 5/10/99

[illegible] Plant Systems [illegible] 1 [illegible] 1

215005 Average Power Range Monitor/Local Power Range Monitor System

A1. Ability to predict and/or monitor changes in parameters associated with operating the APRM/LPRM controls including:

A1.07 APRM (gain adjustment factor) 3.0 3.4

[illegible] a. - not procedurally directed b. - adjustment required if 2% different from calculated power c. - allowed to go to 1.02 d. - correct answer

Reference Title	Facility Reference Number	Section	Page Number	Revision	L.O.
Average Power Range Monitor	SY017 I-4 Fact Sheet	Misc 6.a & b.	3	1	12 & 13
Unit 1 Tech Specs		3.2.4 & 3.3.1.1	3.2-7 & 3.3-3	178	

Material Required for Examination: None

Question Source: New Question Modification Method:

Question Source Comments:

Comment Type	Comment

Record Number: 38 RO Number: SRO Number: 38

[illegible]	Excess flow check valve closure effects on level

Given the following conditions:

- Unit 2 is performing a startup from 180 degrees F
- When the point of adding heat is reached the Unit PCO reports that one of the Wide Range Level indicators has started to lower at a slow but steady rate
- This trend continues as the plant heatup continues
- Drywell pressure and temperature are unchanged
- All other level indicators are steady

This level indicator lowering is caused by:

- the instrument d/p cell equalizing valve is leaking by.
- instrument reference leg outgassing occurring.
- a rise in Reactor Building ambient temperatures.
- the instrument reference leg excess flow valve is closed.

[illegible]	d	[illegible]	S	[illegible]	Comprehension	[illegible]	Susquehanna	[illegible]	5/10/99

Tier:	Plant Systems	RO Group:	1	SRO Group:	1

216000	Nuclear Boiler Instrumentation		
K3.	Knowledge of the effect that a loss or malfunction of the NUCLEAR BOILER Instrumentation will have on following:		
K3.24	Vessel level monitoring	3.9	4.1

[illegible]	Level detectors setup such that "0" d/p is high level, isolated reference leg (at 0 psig) will produce increasing d/p during the heatup, thus indicated level lowering a. - would give rising level b. - would result in Notching, only occurs on depressurization less than 450 psig c. - affects reference and variable legs equally d. - correct answer

[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Reactor Vessel Instrumentation	SY017 J-2	IV.A.2.g & Figure 4A	21	2	2 & 4.g

Material Required for Examination	None

Question Source:	New	Question Modification Method:	

Question Source Comments:	

Comment Type	Comment

Record Number:	39	RO Number:		SRO Number:	39

RCIC suction sources on initiation signal

Given the following conditions:

- Unit 2 is operating at 75% power
- The Reactor Core Isolation Cooling (RCIC) system is in a normal standby lineup except that the Pump Suction From CST Valve (F010) has just been closed for a stroke test
- While the F010 is closed RCIC receives a valid initiation signal

Selected the expected RCIC system response to these conditions?

- The Steam To RCIC Turbine Valve (F045) will not open due to pump low suction pressure.
- The Pump Suction From Suppression Pool Valve (F031) will open allowing RCIC to start and inject normally.
- RCIC will start, run up to an overspeed condition and then trip.
- The Pump Suction From CST Valve (F010) will open allowing RCIC to start and inject normally.

d | Exam Level: S | Cognitive Level: Application | Facility: Susquehanna | ExamDate: 5/10/99

Plant Systems | 1 | 1

217000 | Reactor Core Isolation Cooling System (RCIC)

A3. | Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including:

A3.01 | Valve operation | 35 | 3.5

a. - F045 will open on initiation signal b. - logic goes to the CST suction first if both valve are closed c. - would trip on low suction pressure if it did start with no suction path d. - correct answer

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
Reactor Core Isolation Cooling System	SY017 C-5	V.A.4.f	15	1	8, 16.c & d

None

Question Source: New

Question Source Comments:

40 | | 40

[illegible] RCIC response to loss of oil pressure while operating

Given the following conditions:

- Unit 2 has experienced a loss of off-site power
- The Reactor Core Isolation Cooling (RCIC) system automatically initiated as designed
- The Extra PCO has placed the RCIC Flow Controller in "Manual" to control flow at 350 gpm to maintain reactor water level
- **While in these conditions a failure of the shaft driven lube oil pump results in a total loss of oil pressure (reading 0 psig)**

Which of the following describes the expected response of RCIC?

- RCIC will immediately trip on low lube oil pressure.
- **RCIC decelerate as the governor valve strokes closed.**
- RCIC speed will remain constant until turbine bearing damage begins.
- RCIC will accelerate and trip on overspeed.

[illegible]	d	[illegible]	S	[illegible]	Comprehension	[illegible]	Susquehanna	[illegible]	5/10/99
[illegible]	Plant Systems	RO Group:	1	SRO Group:	1				

217000 **Reactor Core Isolation Cooling System (RCIC)**

A4. Ability to manually operate and/or monitor in the control room:

A4.01 **RCIC turbine speed** 3.7 3.7

[illegible]: Loss of oil pressure results in RCIC governor valve going full open (spring open, hydraulically closed), **turbine speed increases until trip setpoint** a. - no such RCIC trip b. - **governor valve goes full open on** loss of oil pressure c. - governor valve full open on loss of oil pressure, this one may be chosen if flow controller in "manual" is considered d. - correct answer

[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Reactor Core Isolation Cooling System	SY017 C-5	IV.B.2.g	9	1	6.d & 8

[illegible]: None

Question Source: New

Question Modification Method:

[illegible]:

Comment Type	Comment

[illegible]: 41 [illegible]: [illegible]: 41

	Loss of power affects on ADS logic

Given the following conditions:

- The Automatic Depressurization System (ADS) Manual Initiation pushbuttons "A" and "C" (HS30A and HS30C) have been armed and pressed
- There is no response from the ADS safety relief valves

Which of the following electrical bus failures caused this system response?

- A loss of 250 VDC Bus 1D254
- A loss of 480 VAC Bus 1B210
- A loss of 125 VDC Bus 1D614.
- A loss of 120 VAC Bus 1Y216

	c		S		Memory		Susquehanna	5/10/99

Plant Systems		1		1

218000	Automatic Depressurization System		
K2.	Knowledge of electrical power supplies to the following:		
K2.01	ADS logic	3.1	3.3

ADS Logics are powered by 125VDC. Only one 125VDC bus listed a. - 250 VDC b. - 480 VAC c. - correct answer d. - 120 VAC

Reference Title					
Automatic Depressurization And Overpressure Protection Systems	SY017 C-4	III.E.6.a.	13	1	6.h

None

New

Record Number: 42 RO Number: SRO Number: 42

Question Topic: Remote Shutdown Panel SRV vs ADS operation

Given the following conditions:

- Control Room conditions are such that an evacuation is required
- At the Remote Shutdown Panel, ALL Safety Relief Valve (SRV) Emergency Transfer Switches have been placed in "Emergency"
- Valid Automatic Depressurization System initiation signals and conditions are then received
- No Operator actions are taken

Select the expected automatic SRV response for these conditions.

- Three SRVs will open.
- Six SRVs will open.
- Only the transferred SRVs will open.
- No SRVs will open

Answer:	b	Exam Level:	S	Cognitive Level:	Comprehension	Facility:	Susquehanna	ExamDate:	5/10/99

Tier:	Plant Systems	RO Group:	1	SRO Group:	1

218000	Automatic Depressurization System		
K5.	Knowledge of the operational implications of the following concepts as they apply to AUTOMATIC DEPRESSURIZATION SYSTEM:		
K5.01	ADS logic operation	3.8	3.8

Explanation of Distracters: SRVs "A", "B" and "C" are transferred to RSP, does not affect ADS SRVs or their operation a. - 6 ADS SRVs will open b. - correct answer c. - are not ADS SRVs d. - normal ADS operation for these conditions

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
Automatic Depressurization And Overpressure Protection Systems	SY017 C-4	III.A.4.c.(3) & F.3.	5 & 16	1	7 & 10

Material Required for Examination: None

Question Source: Previous 2 NRC Exams **Question Modification Method:** Editorially Modified

Question Source Comments: SSES NRC Exam 04/96 - changed to bullet format, added amplifying conditions

Comment Type	Comment

Sequence Number: 43 **RO Number:** **SRO Number:** 43

[illegible] Actions for Inop MSIV

Given the following conditions:

- Unit 1 is operating at 50% power
- Main Steam Isolation Valve (MSIV) stroke testing is in progress
- The Inboard MSIV in the "A" steam line did not fully stroke closed

Select the required actions.

- Verify the "A" steam line Outboard MSIV is operable and continue plant operation indefinitely.
- Close and deactivate the "A" steam line Outboard MSIV and continue plant operation indefinitely.
- Close and deactivate the "A" steam line Outboard MSIV within 8 hours and commence a shutdown to be in Mode 4 in 36 hours.
- Verify the "A" steam line Outboard MSIV is operable within 8 hours and commence a shutdown to be in Mode 4 in 36 hours.

[illegible] b [illegible] S [illegible] Application [illegible] Susquehanna [illegible] 5/10/89

[illegible] Plant Systems RO Group: 1 SRO Group: 1

223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off

K4. Knowledge of PCIS/NSSSS design feature(s) and/or interlocks which provide for the following:

K4.05 Single failures will not impair the function ability of the system 2.9 3.1

[illegible] a. - TS requires closing the other MSIV, but can still operate b. - correct answer c. - not required to shutdown d. - TS requires closing the other MSIV but not to shutdown

[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
nit 1 Tech Specs		3.6.1.3	3.6-8	178	
Primary Containment Isolation	SY017 E-3			1	11

[illegible] Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

[illegible] New [illegible]

[illegible]

Document Type	Comment

Record Number: 44 RO Number: SRO Number: 44

Use of fuel Hoist Override during core alts.

Given the following conditions:

- Unit 2 is shutdown with core alterations in progress
- While a fuel bundle is being raised out of the core the "Normal Up" light illuminates and the fuel hoist stops
- The Fuel Grapple position indicator (Z) reads 20
- The expected "Normal Up" position should be 16

Which of the following describes the use of the "Hoist Override" pushbutton for these conditions?

- Hoist Override may be used to raise the grapple only to the "Normal Up" position of 16 with Refueling SRO explicit permission.
- With the refueling bridge over the core, the Hoist Override pushbutton is bypassed and is unavailable for use.
- With irradiated fuel on the hoist use of the Hoist Override pushbutton is procedurally prohibited.
- Hoist Override may be used for raising the grapple one "Z" direction increment at a time if a second licensed operator is available for concurrent position verification.

	c		S		Memory	Facility:	Susquehanna		5/10/99

	Plant Systems		3	SRO Group:	2

234000	Fuel Handling Equipment		
K5.	Knowledge of the operational implications of the following concepts as they apply to FUEL HANDLING EQUIPMENT:		
5.02	Fuel handling equipment interlocks	3.1	3.7

Explanation of Answer: a, b, & d. - Hoist override NEVER allowed to be used with irradiated fuel or components on the hoist unless going into a fuel sipping canister c. - correct answer

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	LO
Fuel Handling System Information Sheets (Unit 2)	SY017 M-2	IV.B.2.c.4).b).(2)	32	3	8.c & 11

Material Required for Examination: None

Question Source: Previous 2 NRC Exams

Question Modification Method: Significantly Modified

Question Source Comments: SSES NRC Exam 04/96 - changed stem to bullet format with additional information, two new distractors

Comment Type	Comment

Record Number: 45 RO Number: SRO Number: 45

MSIV operator actuator design

Given the following conditions:

- Unit 1 is performing a reactor startup and heatup
- The reactor is critical and pressure is 150 psig
- Instrument air was lost to the Outboard Main Steam Isolation Valves (MSIV) and they drifted closed
- All expected automatic actions occurred but NO operator actions were taken
- Instrument air has just been restored and the air header has repressurized

Which of the following is the expected response of the Outboard MSIVs and the reason for that response?

The MSIVs will:

- reopen as soon as instrument air has repressurized the lines, the accumulators and the valve actuators.
- remain closed until the control switches are placed in "Close" and the NSSSS Isolation reset push-buttons (Div I and II) are pressed.
- reopen as soon as both of the valve pneumatic control solenoids on each MSIV are reenergized.
- remain closed because the differential pressure across the valve will prohibit opening without equalization.

Answer:	a	Exam Level:	S	Cognitive Level:	Comprehension	Facility:	Susquehanna
Exam Date:	5/10/99						

Tier:	Plant Systems	RO Group:	2	SRO Group:	3

239001	Main and Reheat Steam System		
K6.	Knowledge of the effect that a loss or malfunction of the following will have on the MAIN AND REHEAT STEAM SYSTEM:		
K6.02	Plant air systems	3.2	3.2

Explanation of Answer: a. - correct answer, with switches in "Auto" and d/p less than 150 psid, repressurizing the lines should result in MSIVs reopening b. - no NSSSS isolations occurred c. - solenoids should still be energized d. - d/p should be minimized but MSIVs will open to at least 200 psid

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	LO
Main Steam System	SY017 H-2	IV.A.4.e, VI.C & Figure 6	7,8 & 24	1	3 & 6.c

Material Required for Examination: None

Question Source: New

Question Modification Method:

Question Source Comments:

Comment Type	Comment

[redacted] 46 [redacted] [redacted] 46

[illegible]: Failed SRV vacuum breaker indications

Given the following conditions:

- Unit 2 has experienced a closure of all Main Steam Isolation Valves from 100% power
- Reactor pressure control is via manual Safety Relief Valves (SRV) operation to maintain pressure less than 965 psig

Which of the following is a direct indication that both of the SRV discharge line vacuum breakers on an SRV have failed "open" for these conditions?

- SRV tail pipe temperatures are abnormally high for current plant conditions.
- Plant parameter limits requiring RPV Flooding may be reached sooner then anticipated.
- The Suppression Chamber to Drywell vacuum breakers are cycling each time the SRV is opened and then closed.
- Plant parameters may exceed the Heat Capacity Temperature Limit curve earlier than expected.

[illegible]	b	[illegible]	S	[illegible]	Comprehension	Facility:	Susquehanna	[illegible]	5/10/99
[illegible]	Plant Systems	[illegible]	1	[illegible]	1				

239002	Relief/Safety Valves		
K5.	Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES:		
K5.06	Vacuum breaker operation	2.7	3.0

...tion of Answer: a. - tailpipe temps will remain constant or decrease for these conditions b. - correct answer, reach RPV Sat Curve limits sooner as DW temps/Instrument run temps increase c. - SRV vacuum breakers discharge to drywell, Supp Chamber to Drywell vacuum breakers will not open d. - little energy going to the supp pool, no level or temp change

[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
Primary Containment Control Flowchart	EO-100-103	Step DW/T-3 & Fig 1		0	
Automatic Depressurization And Overpressure Protection Systems	SY017 C-4	III.B.3	6	1	2

Material Required for Examination: None

Question Source: NRC Exam Bank

Question Modification Method: Editorially Modified

[illegible]: VY NRC Exam 01/99 - changed stem to bullet format, modified distractors, new correct answer to fit SSES plant specific information

[illegible]	[illegible]

Record Number: 47 RO Number: SRO Number: 47

Load reject circuits/Intercept Valve fast closure

Which of the following describes how the main turbine is protected from overspeed conditions when a load reject occurs at 30% power? (See attached figure.)

The Electro-Hydraulic Control (EHC) system:

- fast acting solenoids will initiate a fast closure of the Intercept Valves.
- power/load unbalance circuit will initiate a fast closure of the Turbine Control Valves.
- speed and acceleration circuit will throttle the Intercept Valves closed.
- power/load unbalance circuit will throttle the Turbine Control Valves closed.

a | S | Comprehension | Susquehanna | 5/10/99

Plant Systems | 1 | 1

241000 Reactor/Turbine Pressure Regulating System

K6. Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR/TURBINE PRESSURE REGULATING SYSTEM:

K6.20 Main generator | 2.8 | 3.0

a. - correct answer, load reject circuit arms at 20% but only fast closes the IV b. - PLU circuit does not arm until 40% c. - load reject circuit fast closes the IV d. - PLU does not throttle the TCV

EHC Pressure Control & Logic	SY017 A-8	III.A.4.g & Figure 11	24-26	0	6.b & 8.g

Figure 8 from SY017 A-8 EHC Logic Diagram

New

48 | | 48

A 400 SHT.5:6 REV.2 DATE 11/30/85

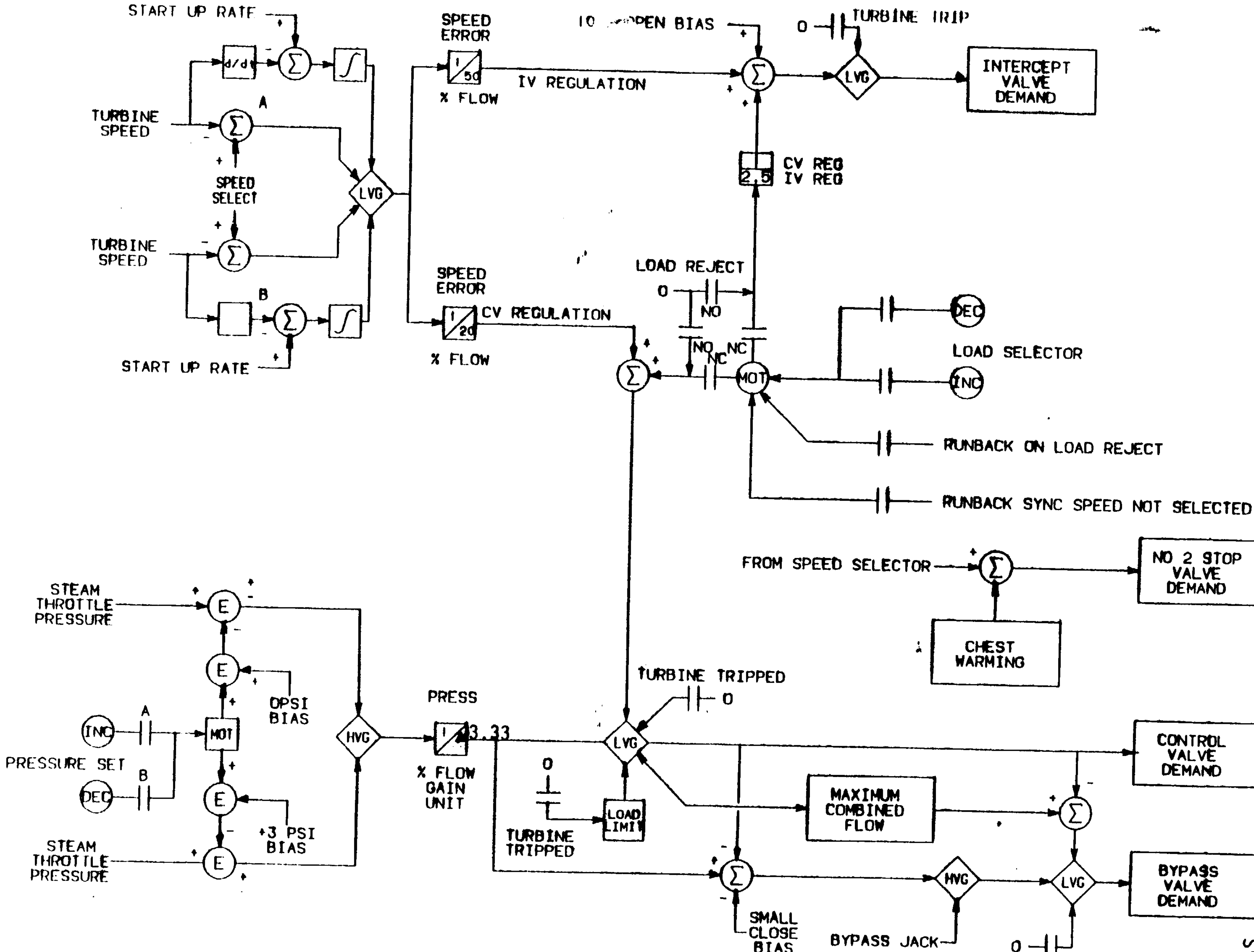


FIGURE 8
SPEED AND ACCELERATION CONTROL UNIT

SY017 A-8

Exceeding 1st stage pressures during shell warming

During a Unit 1 startup and heatup in accordance with GO-100-002, "Plant Startup, Heatup And Power Operation", the operator is directed to maintain turbine first stage pressure less than 120 psig during shell warming.

Which of the following would be expected to occur if this value is exceeded?

- Main turbine Exhaust Hood Spray will initiate.
- Reactor scram.
- Main turbine overspeed trip
- Main Steam Isolation Valve closure.

b | S | Comprehension | Susquehanna | 5/10/99

Plant Systems | 2 | 2

245000 Main Turbine Generator and Auxiliary Systems

A1. Ability to predict and/or monitor changes in parameters associated with operating the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS controls including:

A1.07 First stage turbine pressure | 2.8 | 2.8

a. - Spray cools the LP turbine exhaust boots, shell warming has no impact b. - correct answer, excessive pressure (>123.3 psig) will enable the TSV/TCV reactor scrams c. - if turbine rolls off turning gear will be limited to about 100 rpm by TCV d. - not enough MSL flow to pickup MSIV closure, low pressure closure bypassed by RMS out of "Run"

Plant Startup, Heatup And Power Operation	GO-100-002	6.50	31	31	
Main Turbine Construction	SY017 A-1			0	4 & 5

None

New

40 | | 40

Setpoint setdown operation

Given the following conditions:

- Unit 1 experienced a reactor scram from 95% power
- Reactor water level reached 0 inches
- Feedwater level control remained in "Automatic" and reactor water level currently at +5 inches and is rising
- The Unit PCO has pressed the "Level Setpoint Setdown" pushbutton (HS-C32-1S08)
- All plant systems responded as designed

Reactor water level will:

return to +35 inches.

stabilize at +5 inches.

rise to +13 inches

stabilize at +18 inches

a		S		Memory	Facility:	Susquehanna		5/10/99	
Plant Systems		ID Group:	1	SRO Group:	1				

259002	Reactor Water Level Control System		
A4.	Ability to manually operate and/or monitor in the control room:		
A4.10	Setpoint setdown reset controls: Plant-Specific	3.1	2.9

a. - correct answer, resetting setdown with FWLC in "Auto" will restore level to normal band b. - If less than 18 inches, level would rise to that level then continue to 35 upon reset c. - low level scram, Setpoint Setdown actuation level d. - level that setpoint setdown will go to upon actuation

Reactor Feedwater System	SY017 D-3	V.C.2.s.	36	0	8.f

None

Previous 2 NRC Exams

Concept Used

SSES NRC Exam 04/96 - simplified the stem and distractors, 2 new distractors, modified correct answer

50 | | 50

SGTS operation during a LOCA

Given the following conditions:

- Both Units are operating at 100% power
- The Standby Gas Treatment System (SGTS) is in a normal, standby lineup
- A valid Unit 1 SGTS system initiation signal on high drywell pressure is received

Which of the following is the location from which SGTS will AUTOMATICALLY take suction for these conditions?

SGTS will begin to process the:

- High Pressure Coolant Injection Barometric Condenser Vacuum Pump discharge.
- drywell atmosphere via the drywell and suppression chamber purge connections.
- discharge flow from the running Reactor Building Ventilation Exhaust Fans.
- drywell atmosphere via the drywell vent connections.

	a		S		Memory		Susquehanna	5/10/99
	Plant Systems		1		1			

261000	Standby Gas Treatment System
K1.	Knowledge of the physical connections and/or cause- effect relationships between STANDBY GAS TREATMENT SYSTEM and the following:
1.06	High pressure coolant injection system: Plant- Specific — 3.0 / 3.1

a. - correct answer, HPCI and SGTS start on high drywell pressure b. - receive isolation signal on high drywell pressure c. - suction is from the supply plenum & fans are tripped d. - receive isolation signal on high drywell pressure

Standby Gas Treatment System	SY017 L-3	III.B.2	3	2	11.b & 13

None

New

51		51

[redacted] ESS Bus transfers

Given the following conditions:

- Both Units are operating at 100% power
- All Startup Bus power sources are available
- All four Diesel Generators are available
- The Normal Source Breaker (1A201-01) to ESS Bus 1A is opened with its handswitch on Panel 653
- No other operator actions were taken

Which of the following describes what must occur to reenergize the ESS 1A Bus assuming all systems operate as designed?

- The "A" Diesel Generator will start and the Emergency Source Breaker (1A201-04) will automatically close.
- The Alternate Source Breaker (1A201-09) will automatically close.
- The PCO will have to start the "A" Diesel Generator and close the Emergency Source Breaker (1A201-04).
- The PCO will have to close the Alternate Source Breaker (1A201-09).

[redacted] b [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Plant Systems [redacted] 2 [redacted] 1

262001 A.C. Electrical Distribution

A1. Ability to predict and/or monitor changes in parameters associated with operating the A.C. ELECTRICAL DISTRIBUTION controls including:

A1.05 Breaker lineups 3.2 3.5

[redacted] Bus logic and the logic for the three breakers will always reenergize the bus under these conditions a. - only if the alternate breaker does not close. b. - correct answer c. - only if alternate breaker does not close d. - will auto close

4.16KV/480 VAC ESS Distribution	SY017 G-5C	III.D, E, & F	8-16	1	27.a

[redacted] None

Question Source: New Question Modification Method:

Question Source Comments:

[redacted] 52 [redacted] [redacted] 52

Inop battery Tech Specs

The Unit 1 Unit Supervisor is reviewing the following surveillance data on the 125 VDC 1D640 Battery with the Unit at 100% power:

- Maximum pilot cell specific gravity - 1.215
- Minimum pilot cell specific gravity - 1.205
- Maximum battery cell specific gravity - 1.217
- Minimum battery cell specific gravity - 1.180
- Average battery cell specific gravity - 1.208
- Electrolyte levels in all cells are within limits
- Float voltages in all cells are within limits

What is MAXIMUM permissible time Unit 1 may remain at power for these conditions?

2 hours

12 hours

14 hours

31 days

	c		S		Application		Susquehanna		5/10/99

	Plant Systems		2		2

263000 D.C. Electrical Distribution

2.1 Conduct of Operations

1.12 Ability to apply technical specifications for a system. 2.9 4.0

a. - time allowed to restore a subsystem b. - time to get to Mode 3 if cannot restore subsystem c. - correct answer, with specific gravity on low cell more than 0.020 below average, must declare the battery inop immediately (3.8.6), 3.8.4.A & B give 2 hours to restore and 12 to get to Mode 3 d. - 3.8.6.A limit to restore Cat A and B limits

Unit 1 Tech Specs		3.8.4 & 3.8.6	3.8-23 - 28 & 3.8-32 - 36	178	
125 Volt DC Distribution	SY017 G-3			2	17

Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

NRC Exam Bank | Concept Used

SSES NRC Exam 09/96 - changed question from cell voltage to specific gravity question

Record Number: 53 RO Number: SRO Number: 53

[redacted] Local mode operation of the Diesel Generators

At the "C" Diesel Generator Local Control Panel the Control Mode Select switch has been placed in "Local".

Which of the following describes the operational status of the "C" Diesel Generator?

The "C" Diesel Generator:

[redacted] must be manually started by the local operator on either a loss of off-site power or a LOCA signal.

[redacted] will automatically start on a loss of off-site power but will not respond to a LOCA signal.

[redacted] automatically starts on a LOCA signal but must be manually stared by the local operator on a loss of off-site power.

[redacted] will automatically start in response to both a loss of off-site power and a LOCA signal.

[redacted] a [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Plant Systems [redacted] 1 [redacted] 1

264000 Emergency Generators (Diesel/Jet)

A3. Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including:

A3.01 Automatic starting of compressor and emergency generator 3.0 3.1

[redacted] b, c, & d. - Local operation bypasses the LOCA and LOOP start signals a. - correct answer

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Diesel Generators	OP-024-001	3.5.8 Note	39	33	
Diesel Generators	SY017 G-1			3	15.b

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]

[redacted] 54 [redacted] [redacted] 54

[redacted] MSL Rad Monitor vs MSIV closures

Given the following conditions:

- Unit 2 is operating at 55% power with the "C" Main Steam Line (MSL) isolated (Inboard and Outboard MSIVs are closed)
- A fuel failure results in rising main steam line radiation levels

Which of the following describes the Main Steam Line Radiation Monitor automatic MSIV closure functions under these conditions?

- With the "C" MSL MSIVs closed, the "C" MSL Rad Monitor signal is removed from the circuitry and the isolation logic is modified to a one-out-of-three to close the remaining 6 MSIV.
- The physical location of the 4 MSL Rad Monitors allows each of them to "see" all four steam lines providing for a normal MSIV closure based upon rad levels in the operating steam lines.
- With the "C" MSL MSIVs closed, the "C" MSL Rad Monitor will have a "downscale" signal present providing one of the two required "trips" for the isolation.
- The physical location of the 4 MSL Rad Monitors upstream of the Inboard MSIV provides for continued monitoring of the "C" MSL even though it is isolated.

[redacted] b [redacted] S [redacted] Comprehension [redacted] Susquehanna [redacted] 5/10/99

[redacted] Plant Systems [redacted] 2 [redacted] 2

272000 Radiation Monitoring System

K4. Knowledge of RADIATION MONITORING System design feature(s) and/or interlocks which provide for the following:

K4.01 Redundancy 2.7 2.8

[redacted] a. - no direct correlation between MSIV position and Rad Monitor isolation logic b. - correct answer, MSL Rad Monitors are more "area" rad monitors than "process" rad monitors c. - "C" Rad Monitor will still be seeing radiation from the other 3 lines, no downscale possible d. - located downstream of outboard MSIVs

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Process Radiation Monitoring System	SY017 B-2	IV.C.1.a	10	0	3 & 6

[redacted] None

Question Source: New [redacted]

Question Source Comments:

[redacted]	[redacted]

[redacted] 55 [redacted] [redacted] 55

[redacted] SGTS operability vs fire suppression

Given the following conditions:

- Unit 1 is operating at 90% power
- The functional test of the Standby Gas Treatment (SGTS) Fire Suppression System has just been performed
- The test results were UNSATISFACTORY for both SGTS trains

How does this failure impact continued plant operation?

[redacted] Both trains of SGTS are Inoperable requiring the plant to be in Mode 3 within 16 hours.

[redacted] The secondary containment is Inoperable requiring the plant to be in Mode 3 within 16 hours.

[redacted] Both trains of SGTS remain Operable with no restrictions on plant operations.

[redacted] A continuous fire watch is required in the area when running either SGTS train.

[redacted] c [redacted] S [redacted] Application [redacted] Susquehanna [redacted] 5/10/99

[redacted] Plant Systems [redacted] 2 [redacted] 2

286000 Fire Protection System

K3. Knowledge of the effect that a loss or malfunction of the FIRE PROTECTION SYSTEM will have on following:

K3.03 Plant protection 3.6 3.8

[redacted] Inop SGTS fire suppression does NOT inop SGTS a. - 4 hours to restore one SGTS to Operable or 12 hours to get to Mode 3 b. - 4 hours to restore Sec Ctmt to Operable or 12 to be Mode 3 c. - correct answer d. - TRM requires hourly fire watch

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Unit 1 Tech Specs		3.6.4.3	3.6-42	178	
Unit 1 TRM		3.7.3.2	3.7-8	8/31/1998	
Standby Gas Treatment System	SY017 L-3			2	12

[redacted] Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

[redacted] New [redacted]

[redacted]

[redacted] 56 [redacted] [redacted] 56

[redacted] APRM adjustments with MFLPD greater than RTP

Given the following conditions on Unit 1:

- MFLPD 0.91
- MFLCPR 0.80
- Reactor power 89%
- Core flow 85%

What are the proper actions for these conditions?

[redacted] Reduce the APRM scram setpoints by a multiple of the RTP/MFLPD ratio.

[redacted] Initiate immediate corrective action to restore LHGR to within limits in 1 hour.

[redacted] Reduce the APRM Gain Adjustment Factor by the ratio of MFLPD/RTP.

[redacted] Initiate immediate corrective action to restore MCPR to within limits in 1 hour.

[redacted] a [redacted] S [redacted] Application [redacted] Susquehanna [redacted] 5/10/99

[redacted] Plant Systems [redacted] 3 [redacted] 3

290002 Reactor Vessel Internals

A2. Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:

A2.05 Exceeding thermal limits 3.7 4.2

[redacted] a. - correct answer, TS 3.2.4 & COLR Table 7.2-1, RTP/MFLPD is T and is less than "0", T in the formula reduces the scram setpoints b. - action not required, limits for LHGR restoration are 2 hours c. - ratio of MFLPD/RTP is reversed d. - action not required, limits for MCPR restoration are 2 hours.

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
Unit 1 Tech Specs		3.2.4	3.2-7	178	
Unit 1 TRM		3.2 COLR Table 7.2-1	26 & 27	0	
Reactor Vessel And Internals	SY017 J-1			3	2

[redacted] Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

[redacted] New [redacted]

Question Source Comments:

[redacted]

[redacted] 57 [redacted] [redacted] 57

[redacted] Plant conditions when the actions required by ON-164-002 are applicable

With Unit 1 performing a startup from Cold Shutdown when do the operator actions required by Technical Specifications first become applicable should a Recirculation Pump trip occur?

[redacted] The Reactor Mode Switch has been placed in "Run".

[redacted] The reactor is at or above criticality.

[redacted] The Reactor Mode Switch has been placed in "Startup/Hot Standby".

[redacted] Reactor coolant temperature is > 200 degrees F.

[redacted] c [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 2

295001 Partial or Complete Loss of Forced Core Flow Circulation

AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:

AA1.01 Recirculation system 3.5 3.6

[redacted] Per TS 3.4.1, Recirc Loop operability required in Modes 1 and 2 a. - entry into Mode 1 b. - well past Mode 1 c. - correct answer, Mode 2 entry d. - Mode 3 entry

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Unit 1 Tech Specs		3.4.1	3.4-1	178	
Reactor Recirculation System And Motor Generator Set	SY017 L-8			1	48

[redacted] Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 56 [redacted] [redacted] 58

[redacted] Operating in Region II of Power/Flow Map

Given the following conditions:

- Unit 1 had been operating at 90% power
- The "A" Recirculation Pump tripped
- Parameter verification shows the plant operating in Region II of the Power/Flow Map

Select the desired method for exiting this region.

- Raise flow by raising the speed of the "B" Recirculation Pump
- Place the Reactor Mode Switch in "Shutdown" and enter ON-100-101, "Scram"
- Raise flow by restarting the "A" Recirculation Pump.
- Reduce power by reducing recirculation flow.

[redacted] a [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 2

295001 Partial or Complete Loss of Forced Core Flow Circulation

AK1. Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:

AK1.02 Power/flow distribution 3.3 3.5

[redacted] a. - correct answer b. - Not required for Region II, only Region I c. - cannot exit region by restarting tripped pump d. - with current rod line, will not exit Region

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Loss Of Recirculation Flow	ON-164-002	3.3.2 & 3.3.3	3	16	
ower/Flow Map		Stability regions & reqt's			
Reactor Recirculation Control System	SY017 L-9			1	44

[redacted] None

[redacted] NRC Exam Bank [redacted] Significantly Modified

[redacted] River Bend NRC Exam 01/97 - changed stem to bullet format and SSES specific data, one new distractor

[redacted] 59 [redacted] [redacted] 59

[redacted] Reactor scram on low vacuum

Given the following conditions:

- Unit 2 is operating at 22% power with power ascension in progress
- All plant systems are operating as designed
- Main condenser backpressure is 6.0" HgA and is rising from 3.0" HgA
- No operator actions are taken

The reactor will scram due to:

[redacted] a main turbine trip.

[redacted] main steam isolation valve closure.

[redacted] high reactor pressure.

[redacted] low reactor water level.

[redacted] d [redacted] S [redacted] Comprehension [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 2

295002 Loss of Main Condenser Vacuum

AK2. Knowledge of the interrelations between LOSS OF MAIN CONDENSER VACUUM and the following:

AK2.05 Feedwater system 2.7 2.7

[redacted] a. - bypassed until 30% power b & c. - scrams on low level from feed pump first d. - correct answer, feed pumps trip, lowering level picks up scram

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Loss Of Main Condenser Vacuum	ON-143-001	2	2	11	
Condenser Air Removal	SY017 D-2			1	12

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 60 [redacted] [redacted] 60

RCIC operation during station blackout

Given the following conditions:

- A Station Blackout (SBO) has occurred
- Unit 1 Reactor water level control is via Reactor Core Isolation Cooling (RCIC)
- The Condensate Storage Tank is NOT available due to a tank rupture

Which of the following describes how RCIC operation for level control is accomplished with steadily rising suppression pool temperatures (and subsequent rising lube oil temperatures) during the SBO?

- RCIC suction is lined up to the Refueling Water Storage Tank (RWST) for a source of cool water.
- RCIC lube oil cooling water is supplied from the Fire Protection System.
- The RCIC High Lube Oil Temperature Trip is bypassed when the High Turbine Exhaust Pressure Trip is bypassed.
- RCIC will be run only as necessary to maintain reactor water level +13 to +54".

	b		S		Comprehension		Susquehanna		5/10/99

	Emergency and Abnormal Plant Evolutions		2		1

295003	Partial or Complete Loss of A.C. Power		
AA1.	Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER:		
AA1.03	Systems necessary to assure safe plant shutdown	4.4	4.4

a. - the RWST lineup is a cross-tie to the CST requiring an intact CST b. - correct answer c. - no such trip d. - RCIC is required to be run continuously during a SBO

RCIC Operating Guidelines During Station Blackout	EO-100-033	2.6 & 4.4	4 & 6	8	
Reactor Core Isolation Cooling	SYO17 C-5			1	8 & 10

None

New

61 61

[redacted] Loss of one RPS bus affect on condenser vacuum

Given the following conditions:

- Unit 2 is performing a startup with the Reactor Mode Switch in "Startup/ Hot Standby"
- Main condenser vacuum has been established
- The Outboard Main Steam Isolation Valves have just been opened and steam line warming is in progress
- The "B" Reactor Protection System MG set has just tripped
- The alternate power supply is not available

How will this bus loss affect the plant assuming it is NOT restored as directed by ON-158-001, "Loss Of RPS"?

[redacted] Main condenser vacuum will begin to degrade.

[redacted] The Recirculation Pumps will immediately trip.

[redacted] The Scram Discharge Volume will begin filling.

[redacted] The Outboard Main Steam Isolation Valves will begin to drift closed.

[redacted] a [redacted] S [redacted] Application [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 1

295003 Partial or Complete Loss of A.C. Power

AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following:

AK2.04 A.C. electrical loads 3.4 3.5

[redacted] a. - correct answer, for these conditions the Vacuum Pump will be running to maintain vacuum, with a loss of either RPS bus, the pump trips and suction valves close on loss or MSL Rad Monitor input b. - no affect for this power level c. - both RPS buses must deenergize to close SDV Vent and Drain Valves d. - only half isolation signal present

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Loss Of RPS	ON-158-001	14.30 - 14.33	24	4	
Reactor Protection System	SY017 L-5			1	6 & 7

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]

[redacted] 62 [redacted] [redacted] 62

Loss of 125 VDC affect on paralleled diesel

The "D" Diesel Generator is running with its Unit 1 output breaker (!A204-04)closed following a valid start signal. 125 VDC Bus 1D644 is then deenergized.

The "D" Diesel Generator:

- output breaker will trip and the engine may trip on overspeed.
- will trip and the output breaker will have to be opened locally.
- will continue running as before but all automatic protective features are Inoperable.
- should be placed in Local Control Mode and the DC power selector transferred to the Unit 2 power supply.

	c		S		Memory		Susquehanna		5/10/99

	Emergency and Abnormal Plant Evolutions		2		2

295004 Partial or Complete Loss of D.C. Power

AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER:

AA1.02 Systems necessary to assure safe plant shutdown 3.8 4.1

Loss of 125VDC removes all control and start functions a. - does not effect breaker operation b. - no auto trips available c. - correct answer d. - for a loss of DC while DG shutdown

Loss Of 125V DC Bus 1D640	ON-102-640	5.0 Section 1D644	7	2	
Off-Normal Procedures	AD045			4	2 & 3

None

Question Source: New

Question Modification Method:

Record Number: 63 | Number: | Number: 63

Overspeeding turbine actions

Given the following conditions:

- Unit 1 was operating at 100% when a generator fault resulted in a main turbine trip
- The Extra PCO verified the generator and turbine trip but reports that turbine speed is 1920 rpm and is rising

Which of the following should be directed by the Unit Supervisor?

- Trip the Main Turbine at the front standard.
- Open the Moisture Separator Main Steam Cross-Around line drain valves.
- Break main condenser vacuum.
- Close the Main Steam Isolation Valves.

d | S | Memory | Susquehanna | 5/10/99

Emergency and Abnormal Plant Evolutions | 1 | 2

295005 Main Turbine Generator Trip

2.4 Emergency Procedures and Plan

2.4.48 Ability to interpret control room indications to verify the status and operation of system, and understand how operator action s and directives affect plant and system conditions. 3.5 3.8

a. - possible follow-action but not quick enough to protect the turbine b. - drain lines do not have the capacity to rapidly dump all the steam from between HP and LP turbines c. - not procedurally directed d. - correct answer, though a subsequent action these conditions require immediate actions before the ON is entered

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
Scram	ON-100-101	5.3.3.	4	5	
Main Turbine Construction	SY017 A-1			0	17.a

None

New

64 | | 64

[redacted] EO-113 entry from ON-100-101

While operating in accordance with ON-100-101, "Scram", on a normal plant shutdown reactor scram, which of the following criteria is utilized to determine if EO-100-113, "Level/Power Control" entry is also required?

[redacted] The position and number of control rods inserted.

[redacted] The value of reactor Source Range Monitor (SRM) period after rod movement and detector insertion is complete.

[redacted] The status of the Average Power Range Monitor (ARPM) "Downscale" lights.

[redacted] The ability to monitor instrumentation for valid, current reactor power level.

[redacted] a [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/1/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 1 [redacted] 1

295006 SCRAM

AK1. Knowledge of the operational implications of the following concepts as they apply to SCRAM:

AK1.02 Shutdown margin 3.4 3.7

[redacted] a. - correct answer, >1 rod >00, enter EO-113, no EO-102 entry required b. - SRM period provides indication that the reactor is "shutting down" not "shutdown" c. - APRM downscales not in is EO-102 entry (>5% power) d. - power not known is EO-102 entry

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Scram	ON-100-101	7.0	9	5	
Off-Normal Procedures	AD045			0	2

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 65 [redacted] [redacted] 65

ADS operation with TBV already open

Given the following conditions:

- Unit 1 has been scrammed
- A large coolant leak into the drywell is occurring
- In anticipation of rapid depressurization, all Bypass Valves have been opened
- Reactor pressure has been reduced to 175 psig
- Conditions worsen requiring entry into EO-100-112, "Rapid Depressurization"

Select the required actions for these conditions.

- Open the 6 ADS valves and close the Bypass Valves.
- Close the Bypass Valves and open the 6 ADS valves
- Open the 6 ADS valves and leave the Bypass Valves open.
- Complete the depressurization using only the Bypass Valves.

	c	Exam Level	S	Cognitive Level	Comprehension	Facility:	Susquehanna		5/10/99

	Emergency and Abnormal Plant Evolutions		1		1

295007	High Reactor Pressure
AA1.	Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE:
AA1.04	Safety/relief valve operation: Plant-Specific — 3.9 — 4.1

a. & b. - Both the ADS valves and the BPV once opened shall not be closed by the operator c. - correct answer d. - entry into EO-100-112 requires ADS valves opening, no procedural guidance for depress only with BPV

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
RPV Control	EO-000-102	Step RC/P-3	29	0	
Rapid Depressurization	EO-000-112	Step RD-8	7	0	
Emergency Operating Procedures	PP002A			4	7 & 17

Unit 1 EOP Flowcharts with entry conditions removed

New

65 | | 65

[redacted]	RWCU system isolation requirements

Given the following conditions:

- Unit 1 is operating at 100% power
- A loss of coolant accident occurs
- Reactor water level is -50 inches
- Drywell pressure is 2.4 psig
- All plant systems respond as designed

Using the attached Reactor Water Cleanup System diagram, determine the valves that require operator action to be closed for completion of the system isolation for these plant conditions.

[redacted] ~~HV-144-F001 and HV-144-F004~~

[redacted] HV-144-F042 and HV-144-F104

[redacted] HV-14182A and HV-14182B

[redacted] ~~HV-144-F100 and HV-144-F106~~

[redacted]	c	[redacted]	S	[redacted]	Memory	[redacted]	Susquehanna	[redacted]	5/10/99

[redacted]	Emergency and Abnormal Plant Evolutions	[redacted]	1	[redacted]	1

295009	Low Reactor Water Level		
2.4	Emergency Procedures and Plan		
2.4.12	Knowledge of general operating crew responsibilities during emergency operations.	3.4	3.9

[redacted] a. - auto close on low level, no operator action b. - located in drywell, no affect on containment isolation c. - correct answer, required to be closed by operator for low level combined with high drywell pressure to prevent a secondary containment bypass leakage path d. - located in drywell, no affect on containment isolation

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Containment Isolation	ON-159-002	5.0, last paragraph	4	19	
Off-Normal Procedures	AD045			4	3

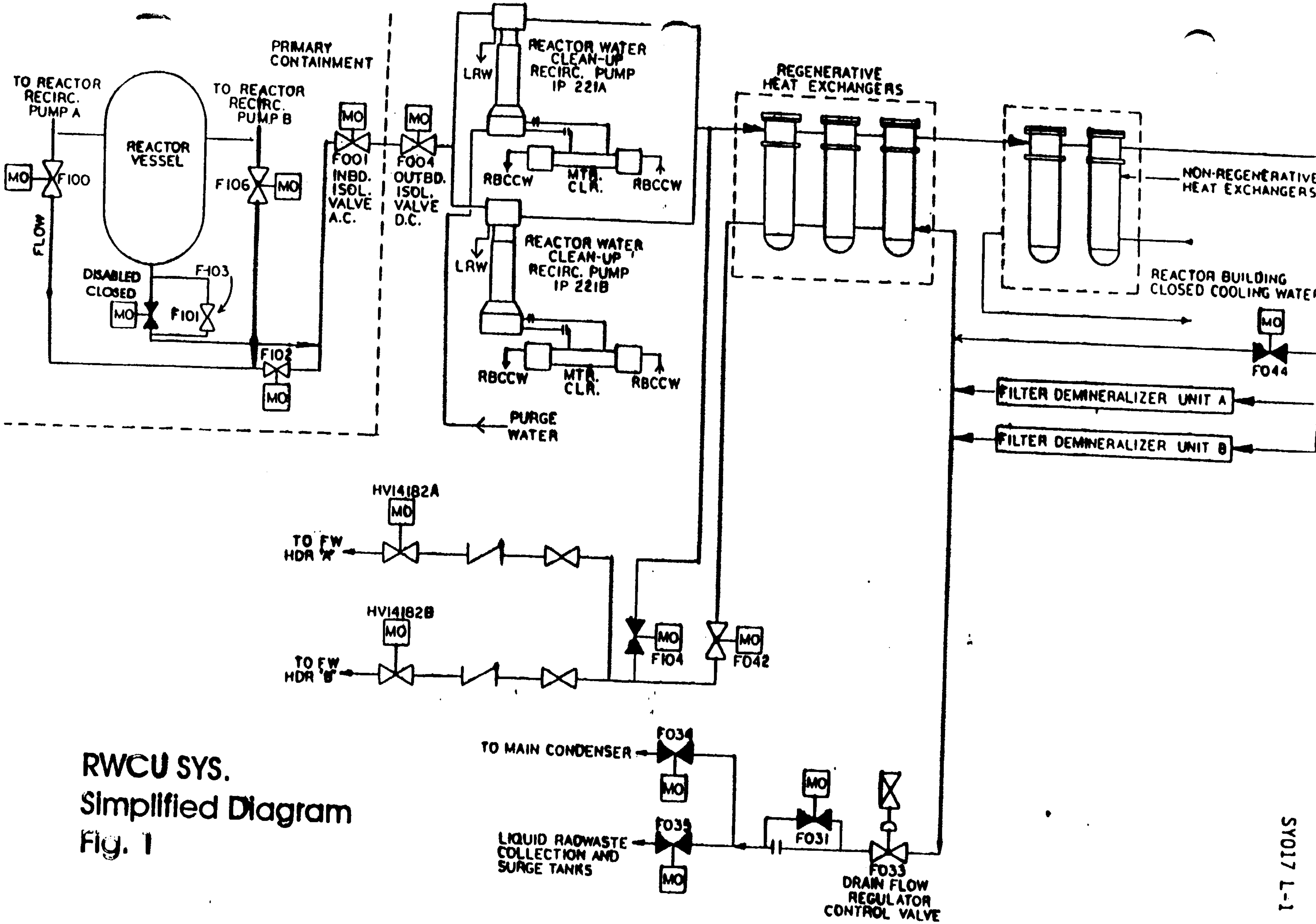
[redacted] Figure 1 from SY017 L-1

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 67 [redacted] [redacted] 67



RWCU SYS.
Simplified Diagram
Fig. 1

SY017 L-1

Reason for reducing CRD flow post scram with no recirc running

Given the following conditions:

- Unit 1 has performed a manual reactor scram as directed by ON-100-101, "Scram"
- The reason for scramming was a trip of both Recirculation Pumps
- The Control Rod Drive Flow Controller has been lowered to "Minimum" as directed by the ON
- The delta T between the reactor bottom head drain and the steam dome is 156 degrees F

For these conditions the operator is required to:

- establish natural circulation flow.
- cooldown to Mode 4.
- start at least one Recirc Pump.
- ensure natural circulation flow will not occur.

	d		S		Memory		Susquehanna		5/10/99

Emergency and Abnormal Plant Evolutions | 1 | 1

295009 Low Reactor Water Level

AK1. Knowledge of the operational implications of the following concepts as they apply to LOW REACTOR WATER LEVEL:

AK1.05 Natural circulation | 3.3 | 3.4

a. - natural circ not desired as thermal stratification is already occurring, this would worsen it b. - not a requirement for these condition, can clear thermal stratification without cooling down c. - not allowed with thermal stratification present d. - correct answer, maintain adequate core cooling but do not startup natural circ flow

Scram	ON-100-101	5.3.11 & 7.0	5 & 9	5	
Off-Normal Procedures	AD045			0	3

None

New

68 | | 68

Drywell and suppression chamber pressure relationship during slow drywell pressure increases

A small Recirculation loop leak exists on Unit 2.

Which of the following describes the expected pressure relationship between the drywell and suppression chamber as the leak continues and pressure rises?

Drywell pressure:

- will rise to about 4.5 to 5.0 psi above suppression chamber, the two pressures will equalize and remain nearly equal as the leak continues.
- will rise to about 0.5 to 1.0 psi above suppression chamber and that differential will be maintained as the leak continues.
- and suppression chamber pressure will initially equalize and maintain that equalization as the leak continues.
- will rise to about 4.5 to 5.0 psi above suppression chamber and that differential will be maintained as the leak continues.

	d		S		Comprehension		Susquehanna		5/10/99

	Emergency and Abnormal Plant Evolutions		1		1

295010 High Drywell Pressure

AK1. Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE:

AK1.01 Downcomer submergence: Mark-I&II | 3.0 | 3.4

a, b & c. - downcomer submergence in the pool is about 11 feet equating to an approximate 4.8 psid between the drywell and the chamber a. - no equalization for these conditions b. - too low a differential, corresponds to the vacuum breakers opening to relieve pressure from the chamber to the drywell c. - drywell and chamber isolated from each other via the downcomer submergence d. - correct answer

Primary Containment Control	EO-000-103	PC/P-1	18	0	
Primary Containment	SY-17 E-2			2	7

None

New

Question Source Comments:

69 | | 69

Drywell Spray Initiation Limit - RHR Pump flow limited on startup

Given the following conditions:

- A large drywell leak has occurred on Unit 1
- Drywell pressure is 28 psig
- Drywell sprays are being started as directed by EO-100-103, "Primary Containment Control"
- When the Inboard Drywell Spray Isolation Valve (F016) is throttled open to establish the required spray flow, the valve strokes to the full open position instead of stopping when the handswitch is released
- No additional operator actions are taken

What is the result of this failure?

- The Residual Heat Removal Pump goes to "runout" and trips on overcurrent.
- Possible drywell spray header damage may occur from water hammer.
- The limits of the RHR & CS Vortex Limit curve may be exceeded damaging the pump.
- Possible drywell damage may occur from exceeding the differential pressure limit.

d | S | Comprehension | Susquehanna | 5/10/99

Emergency and Abnormal Plant Evolutions | 1 | 1

295010 High Drywell Pressure

AK2. Knowledge of the interrelations between HIGH DRYWELL PRESSURE and the following:

AK2.02 Drywell/suppression chamber differential pressure: Mark-I&II | 3.3 | 3.5

a. - max flowrate from this failure is equivalent to normal pump operation, no runout or trip likely b. - not a restriction on spray operation c. - curve max flow is 8000 gpm, pumps will be at 10,000 gpm, curve not applicable d. - correct answer, spray initiation outside of the "old" drywell spray initiation limit curve, still applicable but limiting flow for 30 seconds meets all possible curve conditions

Primary Containment Control	EO-000-103	Step PC/P-7	24	0	
Emergency Operating Procedures	PP002A			4	2 & 7

None

New

70 | | 70

[redacted] Drywell cooling capabilities during a LOCA

Given the following conditions:

- Unit 1 is operating at 100% power
- Drywell temperature and pressure is rising due to a leak
- All expected automatic actions occurred as drywell pressure exceeded 1.72 psig
- EO-100-103, "Primary Containment Control", was entered for high drywell temperature
- ES-134-001, "Restoring Drywell Cooling With A LOCA Signal Present", has been completed

Which of the following describes the current drywell cooling capabilities for these conditions?

The Drywell Cooling Fans are:

[redacted] running with Reactor Building Chilled Water supplying the coolers.

[redacted] running with no cooling water to the coolers.

[redacted] running with Reactor Building Closed Cooling Water supplying the coolers.

[redacted] tripped due to the current LOCA signal.

[redacted] b [redacted] S [redacted] Comprehension [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 2

295012 High Drywell Temperature

K2. Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following:

AK2.02 Drywell cooling 3.6 3.7

[redacted] a & c. - per PCAF, ES-134-001 can no longer be used to bypass and restore RB Chilled Water to drywell coolers if LOCA present b. - correct answer, can start in "slow" only d. - ES-134-001 does allow restart of coolers in "slow" during LOCA

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Restoring Drywell Cooling With A LOCA Signal Present	ES-134-001	4.2	4	9	
Primary Containment Atmosphere Control	SY017 E-6			1	30

Material Required for Examination: None

Question Source: New   Question Modification Method:

[redacted]

[redacted]	[redacted]

Record Number: 71   [redacted] RO Number: 71

Indications of a stuck open SRV

Given the following conditions AFTER a transient from 90% power on Unit 1:

- Reactor power (MWt) is slightly higher
- Generator megawatts (MWe) are slightly lower
- Indicated feedwater flow is greater than indicated steam flow (matched before the transient)
- Reactor water level is slightly higher

These conditions are being caused by:

- isolation of extraction steam to one feedwater heater.
- a stuck open Safety Relief Valve.
- rising main condenser backpressure (degrading vacuum).
- failure of the on-service EHC pressure regulator to a lower output.

b | S | Comprehension | Susquehanna | 5/10/99

Emergency and Abnormal Plant Evolutions | 2 | 1

295013 | High Suppression Pool Temperature

AA1. | Ability to operate and/or monitor the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE:

AA1.02 | Systems that add heat to the suppression pool | 3.9 | 3.9

a. - should not affect level and MWe b. - correct answer, SRV steam bypassing feedwater heating gives lowering feed temps and power increase c. - should not affect level, feed and steam flows d. - backup regulator should control slightly lower pressure

Stuck Open Safety Relief Valve	ON-183-001	1.0	2	17	
Main Steam System	SY017 H-2			1	3

None

New

72 | | 72

Margin to the HCTL while operating SRVs

Given the following conditions:

- Unit 1 has experienced a Main Steam Isolation Valve closure from 100% power
- The control rods did not insert
- EO-100-113, "Level/Power Control", has been entered
- The Safety Relief Valves have been manually opened to control pressure less than 965 psig
- Standby Liquid Control is not available

For these conditions, the Heat Capacity Temperature Limit:

- will steadily become more restrictive.
- will remain constant.
- will steadily become less restrictive.
- has been exceeded.

	a		S		Comprehension		Susquehanna		5/10/99
	Emergency and Abnormal Plant Evolutions		2		1				

295013	High Suppression Pool Temperature		
AK3.	Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE:		
AK3.02	Limiting heat additions	3.6	3.8

b, c & d. - temps increasing, level increasing, reactor pressure decreasing as SLC goes in therefore margin to HCTL gets smaller a. - correct answer

RPV Control	EO-000-102	RC/P-5	32	0	
Emergency Operating Procedures	PP002A			4	2

None

New

73				73

Loss of feedwater heating versus thermal limits

While at 90% power, Unit 1 has experienced a loss of feedwater heating resulting in a feedwater temperature drop of 55 degrees F.

Assuming no operator actions taken, what is the operational concern for these conditions?

- Immediate core flux oscillations
- Recirculation loop jet pump vibrations.
- Violation of the Susquehanna Unit 1 Operating License
- Entry into Region I of the Power/Flow Map

c | S | Memory | Facility: Susquehanna | 5/10/99

Emergency and Abnormal Plant Evolutions | 1 | 1

295014 Inadvertent Reactivity Addition

AK2. Knowledge of the interrelations between INADVERTENT REACTIVITY ADDITION and the following:

AK2.02 Fuel thermal limits | 3.7 | 4.2

a. - not a concern for this power and flow combination b. - increased subcooling does not affect jet pumps c. - correct answer, reactor closer to/may be exceeding Tech Spec/License MCPR limits d. - min flow allowed is 55 mlbm/hr, Region I entry not possible

Loss Of Feedwater Heating Extraction Steam	ON-147-001	5.0	8	10	
Off-Normal Procedures	AD045			0	3

None

New | Question Modification Method:

Record Number: 74 RO Number: SRO Number: 74

[redacted] EOP actions on a scram with RPIS Inop

Given the following conditions:

- Unit 1 is operating at 100% power
- A complete loss of the Rod Position Indication System has occurred requiring a shutdown
- Recirculation flow has been reduced to 55 mlbm/hour and the Reactor Mode Switch placed in "Shutdown"

For these conditions, how will the Unit Supervisor (US) make the determination on whether injection of Standby Liquid Control is required?

[redacted] Control rod position can be verified by demanding an OD-7, Option 1 printout.

[redacted] The Unit PCO can monitor Average Power Range Monitor (APRM) power levels.

[redacted] Control Rod position can be verified by the Rod Worth Minimizer Full Core Display screen.

[redacted] The Unit PCO can verify a red "Scram Valves" light is received for each control rod.

[redacted] b [redacted] S [redacted] Comprehension [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 1 [redacted] 1

295015 Incomplete SCRAM

AA2. Ability to determine and/or interpret the following as they apply to INCOMPLETE SCRAM:

AA2.01 Reactor power 4.1 4.3

[redacted] a. - OD-7 not available with loss of RPIS b. - correct answer, EO-113 ask if power >5% before injecting SLC c. - RWM utilizes RPIS input signal d. - not positive indication the rods have inserted

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
RPIS Failure	ON-155-004	3.5.2.a	6	9	
Level/Power Control	EO-100-113 Sheet 1	Step LQ/Q-3		0	
Emergency Operating Procedures	PP002A			4	4

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 75 [redacted] [redacted] 75

[redacted] Status of Recirc on Control Room Evacuation

A Control Room evacuation is required. All Immediate Operator Actions of ON-100-009, "Control Room Evacuation", were completed prior to leaving.

What will be the current means of core heat removal when the operators arrive to establish control at the Remote Shutdown Panel?

[redacted] Recirculation flow removing core heat for dissipation via the Safety Relief Valves.

[redacted] Natural circulation flow removing core heat for dissipation via Turbine Bypass Valves.

[redacted] Recirculation flow removing core heat for dissipation via the Turbine Bypass Valves.

[redacted] Natural circulation flow removing core heat for dissipation via the Safety Relief Valves.

[redacted] d [redacted] S [redacted] Comprehension [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 1

295016 Control Room Abandonment

AA2. Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT:

AA2.03 Reactor pressure 4.3 4.4

[redacted] a, b, & c. - ON-100-109 directs MSIV closure, tripping RFPs and closing the discharge valves, Recirc Pumps will trip at -38 inches, pressure control on SRVs d. - correct answer

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Control Room Evacuation	ON-100-109	3.1, 4.6 & 4.9.8	2, 9-10 & 12	4	
Off-Normal Procedures	AD045			0	3 & 5

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]

[redacted] 76 [redacted] [redacted] 76

[redacted] Reactor water level control from the Remote Shutdown Panel

Given the following conditions:

- A Unit 1 fire has resulted in the closure of all Outboard Main Steam Isolation Valves from 100% power
- High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) both automatically initiated and are injecting
- The Immediate Operator Actions of ON-100-009, "Control Room Evacuation" were completed
- All Remote Shutdown Panel (RSP) Control Transfer Switches have been placed in "Emergency"
- The RSP operator trips RCIC when reactor water level reaches +54 inches

Reactor water level will:

- lower until RCIC automatically re-initiates at -30 inches.
- lower until HPCI automatically re-initiates at -38 inches.
- lower until both HPCI and RCIC automatically re-initiate.
- continue to rise due to HPCI injection.

[redacted] b [redacted] S [redacted] Application **Facility:** Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 1

295016 Control Room Abandonment

K2. Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following:

AK2.01 Remote shutdown panel: Plant-Specific 4.4 4.5

[redacted] a. - RCIC auto start disabled at RSP b. - correct answer, assuming HPCI not affected by fire, will continue to cycle between -30" and +54" c. - HPCI does restart, RCIC does not d. - HPCI normal +54" trip still in effect

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Control Room Evacuation	ON-100-009	4.6.4 Caution & Att D. 8	10 & 22	4	
Off-Normal Procedures	AD045			0	2 & 3

[redacted] None

Question Source: New **Question Modification Method:**

[redacted]

[redacted]

Record Number: 77 **RO Number:** **SRO Number:** 77

[redacted] Purpose of EO-100-105

Entry into EO-100-105, "Radioactivity Release Control", and completion of the required actions will limit the activity release from:

[redacted] the reactor coolant into the primary containment.

[redacted] the reactor coolant into areas outside the primary and secondary containment.

[redacted] any damage fuel directly into the reactor coolant and plant primary systems.

[redacted] **the reactor coolant into the secondary containment.**

[redacted] b [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 1

295017 High Off-Site Release Rate

AK1. Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE:

AK1.02 Protection of the general public | 3.8 | 4.3

[redacted] *A restatement of the purpose of EO-100-105 a. - taken care of by EO-100-103 b. - correct answer c. -* EO-100-102 d. - EO-100-104

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Radioactivity Release Control	EO-000-105	General	2	0	
Emergency Operating Procedures	PP002A			4	7

[redacted] None

[redacted] New [redacted]

[redacted]

Comment Type	Comment

[redacted] 78 [redacted] [redacted] 78

Limiting parameters on half isolation from loss of RPS Bus

Given the following conditions:

- Unit 2 is operating at 100% power
- The "A" Reactor Protection System (RPS) Bus is on the alternate power supply
- The "B" RPS MG Set has just tripped

Which of the following describes the restrictions on continued plant operation for these conditions?

The plant may operate in Mode 1 for a limited amount of time based upon:

- the availability of the Reactor Building Equipment Drain Sump Pumps.
- the rate at which the instrument air supply to the Outboard MSIVs depressurizes.
- the availability of the Reactor Recirculation Pumps.
- the rate at which the Scram Discharge Volume fills.

c	S	Memory	Susquehanna	5/10/99			

Emergency and Abnormal Plant Evolutions | 2 | 2

295018 Partial or Complete Loss of Component Cooling Water

AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following:

AK2.02 Plant operations | 3.4 | 3.6

a. - Loss of RPS "A" affects drywell equip drain sump pumps b. - loss of one RPS bus gives half isolation but no MSIV closure possible c. - correct answer, loss of RBCCW to drywell (Recirc Pump motor winding coolers), about 7 minutes to high temp alarm d. - loss of one RPS bus does not affect SDV Vent and Drain Valves

Loss Of RPS	ON-158-001	5.0 last paragraph	4	4	
Reactor Protection System	SY017 L-5			1	5 & 5

None

New

79 | | 79

[redacted] When scram required on loss of instrument air

Given the following conditions:

- Unit 1 is operating at 35% power
- Unit 2 Instrument Air is NOT available
- Unit 1 Instrument Air pressure is 105 psig and is slowly lowering

When is Unit 1 REQUIRED to be scrammed?

- More than 2 control rod "drift" alarms are received.
- Instrument Air pressure has reached 95 psig.
- The Scram Discharge Volume high level control rod block is received.
- The red "Scram Valves" light is received for any control rod.

[redacted] a [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 2

295019 Partial or Complete Loss of Instrument Air

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR:

AA2.02 Status of safety-related instrument air system loads (see AK2.1 - AK2.19) 3.6 3.7

[redacted] a. - correct answer, loss of air results in rod drifts, 3 or more drifts require immediate scram b. - 95 psig is the IA - SA crosstie auto opening c. - no procedural guidance for this d. - no procedural guidance for single rod scram

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
ontrol Rod Problems	ON-155-001	3.4.3	10	14	
Loss Of Instrument Air Attachment A	ON-118-001	A	6	11	
Instrument Air	SY017 L-14			0	11

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted] 80 [redacted] [redacted] 80

[redacted]	Single MSIV closure at power effects

With Unit 1 at 75% power, the Inboard Main Steam Isolation Valve (MSIV) in the "A" Main Steam Line fails closed.

Select the expected automatic plant response.

- A half scram on "A" RPS occurs.
- The remaining 7 MSIVs close.
- The reactor will stabilize at a lower pressure.
- The reactor power will stabilize at a higher power.

[redacted]	d	[redacted]	S	[redacted]	Comprehension	[redacted]	Comprehension	[redacted] 5/10/99

[redacted]	Emergency and Abnormal Plant Evolutions	[redacted]	2	[redacted]	2

295020	Inadvertent Containment Isolation		
AK3.	Knowledge of the reasons for the following responses as they apply to INADVERTENT CONTAINMENT ISOLATION:		
AK3.04	Reactor pressure response	4.1	4.1

[redacted]	Single MSIV closure at this power level gives higher reactor pressure (and power) due to increased steam flow head loss, same flow through 3 steam lines now a. - one MSL isolation never causes half scram b. - 75% will not give enough flow through the remaining steam lines to pick up isolation c. - higher pressure is result d. - higher pressure gives higher power

[redacted]					
Main Steam	SY017 H-2	VI.B.1.a & Figure 1	21	1	3 & 5.a

[redacted]	None

[redacted]	New	[redacted]	

[redacted]	

[redacted]	

[redacted]	81	[redacted]		[redacted]	81

[redacted] SRV operation on loss of SDC in Mode 3

Given the following conditions:

- Unit 2 has experienced a loss of Shutdown Cooling while in Mode 3
- Primary and Secondary Containment are established
- The plant has been shutdown for 36 hours
- Reactor pressure is being maintained 20 to 98 psig by opening Non-ADS Safety Relief Valves (SRV) as needed
- Following opening of the "B" SRV, the Extra PCO is unable to close the valve

Which of the following describes the effect of this failure to close as the reactor depressurizes?

[redacted] SRV "Open" position indication from the Acoustic Monitor will be lost as discharge downcomer flow lowers.

[redacted] The reactor will reach saturation temperature with a subsequent reduction in the "time-to-boiling" value.

[redacted] Adequate core cooling will not be maintained for these conditions.

[redacted] The SRV discharge downcomer may begin to reflood with suppression pool water.

[redacted] d [redacted] S [redacted] Application [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 3 [redacted] 2

295021 Loss of Shutdown Cooling

AA1. Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING:

AA1.04 Alternate heat removal methods 3.7 3.7

[redacted] a. - acoustic monitor not used as position indication, flows with SRV open between 20-98 psig may not indicate on monitor anyway b. - reactor already at saturation (Hot Shutdown), >200 degrees F c. - if flow reduction causes core heatup, subsequent pressure increase will raise pressure through open SRV and/or open other SRVs d. - correct answer, inverse of why keep pressure >19 psig

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Loss Of RHR Shutdown Cooling	ON-149-001	3.3.6.a.(2) Note	5	16	
Residual Heat Removal System	SY017 C-1			2	29

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 82 [redacted] [redacted] 82

Loss of CRD actions during a startup

Given the following conditions:

- Unit 1 is performing a reactor startup
- Reactor pressure is 825 psig
- The Reactor Mode Switch is in "Startup/Hot Standby"
- Control Rods 30-15 and 46-47 (both at Notch "00") have accumulator alarms in on low pressure and are being recharged
- The "A" Control Rod Drive Pump is not available
- The "B" Control Rod Drive Pump has just tripped and cannot be restarted
- Charging header pressure has equalized with reactor pressure

Which of the following describes the plant conditions requiring the Reactor Mode Switch be placed in "Shutdown"?

No actions are required to be taken unless:

- charging header pressure cannot be raised to or above 940 psig within 20 minutes.
- control rods 30-15 and 46-47 cannot be returned to Operable status within one hour.
- an accumulator alarm is received on a currently withdrawn control rod.
- control rods 30-15 and 46-47 cannot be returned to Operable status within 20 minutes.

c | S | Memory | Susquehanna | 5/10/99

Emergency and Abnormal Plant Evolutions | 2 | 2

_95022 | Loss of CRD Pumps

AK1. | Knowledge of the operational implications of the following concepts as they apply to LOSS OF CRD PUMPS:

AK1.01 | Reactor pressure vs. rod insertion capability | 3.3 | 3.4

a. - only required if reactor pressure was above 900 psig b. - can remain Inop indefinitely c. - correct answer, meets limits of ON section 3.2.1 d. - time limit for restoring charging header pressure to >940 psig with reactor pressure >900 psig

Loss Of CRD System Flow	ON-155-007	3.2.1	2	13	
Unit 1 Tech Specs		3.1.5	3.1-15 - 17	178	
Control Rod Drive Mechanism	SY017 K-3			3	11

Material Required for Examination: Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

New

Record Number: 83 RO Number: SRO Number: 83

Preventing draining fuel pool to suppression pool during RHR Fuel Pool Cooling Mode

Given the following conditions:

- Unit 1 is in Mode 5
- The "A" Residual Heat Removal (RHR) loop is being placed in the Fuel Pool Cooling mode

Which of the following prevents draining the containment fuel pools to the suppression pool via the RHR Minimum Flow Valve (F007A) when starting the "A" RHR Pump?

- The operator is procedurally directed to establish flow to the fuel pools before the F007A valve automatically opens.
- F007A is manually overridden closed by the operator prior to starting the RHR pump.
- The RHR pump is started with a complete, established flowpath to the fuel pools to prevent this.
- The F007A automatic operation is defeated by lifting leads during the Fuel Pool Cooling mode lineup.

d | S | Memory | Susquehanna | 5/10/99

Emergency and Abnormal Plant Evolutions | 3 | 1

295023 | Refueling Accidents

AK2. | Knowledge of the interrelations between REFUELING ACCIDENTS and the following:

AK2.02 | Fuel pool cooling and cleanup system | 2.9 | 3.2

a. - not directed, valve is disabled b. - no such feature, valve is disabled c. - actual lineup requires valve manipulation following pump start d. - correct answer

RHR Operation In Fuel Pool Cooling Mode	OP-149-003	3.5	6	17	
Residual Heat Removal System	SY017 C-1			2	16 & 24.i

None

New

84 | | 84

Indications of a failed SRV tail pipe in the suppression chamber

Given the following parameters:

- Drywell pressure — 3.5 psig and rising
- Drywell temperature — 145 degrees F and rising
- Suppression chamber pressure — 4.6 psig and rising
- Suppression pool water temperature — 87 degrees F and steady

Which of the following describes what has occurred?

- A downcomer vacuum breaker has failed open during a recirculation leak to the drywell.
- A pipe break into the drywell has occurred with a suppression chamber to drywell vacuum breaker open.
- A safety relief valve tail pipe has broken above the suppression pool water level while the valve is open.
- A recirculation line partial break has occurred with all containment parameters responding as designed.

c	S	Application	Susquehanna		5/10/99

Emergency and Abnormal Plant Evolutions — RO Group: 1 — SRO Group: 1

295024 High Drywell Pressure

EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE:

EA2.04 Suppression chamber pressure: Plant-Specific — 3.9 — 3.9

a. - downcomer vacuum breakers are designed to be open for these conditions, equalize pressure across the drywell floor when drywell pressure less than chamber pressure b. - this would tend to equalize drywell and chamber pressure or even have drywell pressure slightly higher since that is the leak location c. - correct answer, energy into chamber but not into pool, vacuum breakers opening back to drywell when d/p high enough d. - all parameters way too low, especially pool temperature

Primary Containment Control	EO-000-103	PC/P-4	20	0	
Primary Containment	SY017 E-1			2	7 & 13

None

NRC Exam Bank — Significantly Modified

Question Source Comments: Peach Bottom NRC Exam 09/97 - different stem conditions, one new distractor

Comment Type	Comment

85 — 85

SRV tailpipe temp trend during depressurization

Given the following conditions:

- Unit 1 was operating at 100% power
- A severe overpressure transient has resulted in the Safety Relief Valves (SRV) opening in their "Safety Valve" mode
- All valves, with the exception of one, have reseated (closed)
- The required actions of ON-183-001, "Stuck Open Safety Relief Valve" have been completed
- The reactor has been scrammed
- The SRV has NOT closed

As the reactor cools down and depressurizes through the stuck open SRV tail pipe temperature will:

- start at 305 degrees F and will slowly fall following reactor pressure during the depressurization.
- start at 270 degrees F, rise to approximately 300 degrees F and then will slowly fall following reactor pressure during the depressurization below 500 psig
- start at 525 degrees F and will slowly fall following reactor pressure during the depressurization.
- start at 285 degrees F, rise to approximately 325 degrees F and then will slowly fall following reactor pressure during the depressurization below 500 psig

d	S	Comprehension	Susquehanna		5/10/99

Emergency and Abnormal Plant Evolutions	1	1

295025 High Reactor Pressure

K1. Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE:

EK1.03	Safety/relief valve tailpipe temperature/pressure relationships	3.6	3.8

TMI scenario with SRV safety settings about 1200 psig, tailpipe temps start low then rise to peak at about 500 psig then slowly lower, this is a isoenthalpic process from the Mollier diagram NOT the steam tables. A. - approx temp for stuck open SRV at steady pressure b. - temps both too low per Mollier diagram c. - sat temp for 1200 psig d. - correct answer

Automatic Depressurization And Overpressure Protection Systems	SY017 C-4	III.A.3 & F.4	3-4 & 16	1	5
Steam Tables		Mollier Diagram			

Steam Tables - Mollier Diagram

NRC Exam Bank — Editorially Modified

River Bend NRC Exam 01/97 - modified stem and distractors to SSES SRV Safety setpoints

86 86

Startup following high suppression pool temp required scram

The Unit 1 Reactor Mode Switch was placed in "Shutdown" due to suppression pool temperature being greater than the Technical Specification limit.

Suppression pool temperature must be at or below:

- 110 degrees F for 36 hours prior to entering Mode 3.
- 90 degrees F within 24 hours of placing the Reactor Mode Switch in "Shutdown".
- 110 degrees F prior to entering Mode 2 on the ensuring startup.
- 90 degrees F prior to reaching the point of adding heat on the ensuing startup.

d | S | Application | Competence | 5/10/99

Emergency and Abnormal Plant Evolutions | 2 | 1

295026 Suppression Pool High Water Temperature

EA2. Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:

EA2.01 Suppression pool water temperature | 4.1 | 4.2

a. - no requirement to go all the way to Mode 4, once less than 110 can stay in current mode b. - must be less than 90 in 24 hours if operating c. - can be less than 110 up until POAH d. - correct answer, cannot add heat until less than 90 pool temp

Unit 1 Tech Specs		3.6.2.1	3.6-22 & 23	178	
Primary Containment - Structure	SY017 E-1			2	26

Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases

NRC Exam Bank | Concept Used

Hope Creek NRC Exam 08/94 - used idea, modified to SSES Improved Tech Spec numbers

87 | | 87

EOP entry on high supp pool temp during testing

Given the following conditions:

- Unit 1 is operating at 50% power
- Suppression pool cooling is in service
- High Pressure Coolant Injection (HPCI) is operating in the CST to CST mode for a surveillance
- During the surveillance suppression pool temperature reached 96 degrees F

What are the requirements for entry into, and implementation of, EO-100-103, "Primary Containment Control", for these conditions?

- Technical Specifications modify the Emergency Operating Procedure entry condition to 105 degrees F while surveillance testing to the suppression pool is in progress.
- EO-100-103 actions may be deferred for 24 hours while suppression pool temperature is reduced to less than 90 degrees F.
- The HPCI surveillance procedures allow 4 hours to reduce suppression pool temperature below 90 degrees F before EO-100-103 entry is required.
- The actions of EO-100-103 are required to be performed as soon as suppression pool temperature is above 90 degrees F.

	d		S		Comprehension		Susquehanna		5/10/90

	Emergency and Abnormal Plant Evolutions		2		1

295026	Suppression Pool High Water Temperature
2.4	Emergency Procedures and Plan
2.4.1	Knowledge of EOP entry conditions and immediate action steps. 4.3 4.6

a. - Tech Spec actions may be deferred until 105 deg but EOP entry still required b. - EOP entry deferment not allowed c. - no guidance for this provided d. - correct answer

Primary Containment Control	EO-100-103	Entry Conditions		0	
Emergency Operating Procedures	PP002A			4	1.b

Material Provided for Examination: None

Question Source: New | Question Modification Method:

Record Number: 88 | RO Number: | SRO Number: 88

[redacted] Affects of increasing drywell temps on HPCI initiation

Given the following conditions:

- Unit 2 is operating at 100% power
- Drywell pressure and temperatures are rising rapidly
- High Pressure Coolant Injection (HPCI) did not start on high drywell pressure

As these conditions worsen and water level lowers following the scram, HPCI:

[redacted] must be initiated by the operator because wide range level indication will be off-scale low.

[redacted] will initiate late because the wide range level indication will be reading higher than actual water level.

[redacted] will not initiate because wide range level indication will be off-scale high.

[redacted] will initiate early because the wide range level indication will be reading lower than actual water level.

[redacted] b [redacted] S [redacted] Application [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 2

295028 High Drywell Temperature

EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE:

EK1.01 Reactor water level measurement 3.5 3.7

[redacted] a, c & d - as saturation conditions are approached and reaches boiling in the ref and var legs will be seen as lower sensed d/p giving a increasing indicated water level, water level lowering will be lower than wide range seen setpoints a. - level goes high b. - correct answer c. - may eventually reach off-scale high d. - level goes high

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Primary Containment Control	EO-000-103	DW/T-3	39	0	
Emergency Operating Procedures	PP-002A			4	7

[redacted] None

[redacted] NRC Exam Bank [redacted] Editorially Modified

[redacted] Peach Bottom NRC Exam 09/96 - cleaned up stem, made question ECCS system specific

[redacted]	[redacted]

[redacted] 90 [redacted] [redacted] 90

Primary Containment water level measurements to determine if core covered (>TAF)

Which of the following describes how the operator determines if water level in the containment is above the top of active fuel while flooding the primary containment?

Top of active fuel is determined by:

- indicated drywell pressure versus containment level correlation if the drywell is vented to atmosphere.
- a pressure and temperature corrected reading from Wide Range Suppression Pool Level indication.
- a level calculated from the pressure differential between the drywell and the suppression chamber.
- direct reading from the reactor water level Fuel Zone Level indicator if the drywell is vented to atmosphere.

a | S | Memory | Susquehanna | S/7/099

Tier: Emergency and Abnormal Plant Evolutions | RO Group: 2 | SRO Group: 2

295029 High Suppression Pool Water Level

EA2. Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL WATER LEVEL:

EA2.03 Drywell/containment water level | 3.4 | 3.5

a. - correct answer for levels above 64 feet, requires knowing that TAF is 116 feet in containment b. - use this level indicator for levels <49 feet c. - used for 49 to 64 feet d. - Fuel Zone not reliable for these conditions and if it is accurate the drywell is not required to be vented

Primary Containment Water Level Anomaly	ON-159-003	5.3	4	2	
Off-Normal Procedures	AD045			0	2

None

New

90		90

[redacted] Rapid depress during an ATWS

Given the following conditions on Unit 1:

- A failure-to-scram (ATWS) condition exists
- Reactor power is 22%
- Standby Liquid Control is injecting
- The Scram Discharge Volume did NOT isolate
- Suppression pool level is 15 feet and lowering
- A greater than Max Safe Water Level exists in two (2) Reactor Building areas

Which of the following are the appropriate actions for these conditions?

[redacted] Immediately open 6 ADS Safety Relief Valves.

[redacted] Take no action until power is less than 5% or all rods are inserted.

[redacted] Immediately open the Turbine Bypass Valves.

[redacted] Take no action until suppression pool reaches 12 feet.

[redacted] a [redacted] S [redacted] Application [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions RO Group 2 [redacted] 1

295030 Low Suppression Pool Water Level

EA2. Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL:

EA2.01 Suppression pool level 4.1 4.2

[redacted] a. - correct answer, cannot Rapid Depress or use bypass valves with ATWS on low pool level but CAN on two areas > Max Safe Water Level with primary system discharging b. - true but rapid depress should have been done already c. - can't use with ATWS d. - true but rapid depress should have already been done

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Primary Containment Control	EO-000-103	Step SP/L-8	15	0	
Emergency Operating Procedures	PP002A			4	6 & 7

[redacted] Unit 1 EOP Flowcharts with entry conditions removed

Question Source: New Question Modification Method:

[redacted]

Comment Type	[redacted]

[redacted] 91 [redacted] [redacted] 91

Consequences of running RCIC with low suppression pool level

Given the following conditions:

- Reactor Core Isolation Cooling (RCIC) is providing injection to the reactor
- Reactor pressure is 455 psig and lowering
- Suppression pool water level is 16 feet and lowering
- Suppression pool temperature is 155 degrees F and rising
- The plant is operating in accordance with EO-100-103, "Primary Containment Control"

Which of the following is the expected result with RCIC continuing to run under these conditions?

- The Heat Capacity Temperature Limit will be exceeded.
- RCIC will trip.
- Suppression chamber design pressure will be exceeded.
- RCIC will cavitate.

b | S | Application | Susquehanna | 5/10/99

Emergency and Abnormal Plant Evolutions | 2 | 1

295030 | Low Suppression Pool Water Level

EK3. Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL:

EK3.03 RCIC operation: Plant-Specific | 3.6 | 3.7

a & c. - not possible, RCIC exhaust uncovered pressurizing the chamber, RCIC trips on high exhaust pressure b. - correct answer, this is why RCIC is not isolated when HPCI is d. - any vortex limits for RCIC Pump will not be reached

Primary Containment Control	EO-000-103	Step SP/L-6	13	0	
Emergency Operating Procedures	PP002A			4	7 & 9

None

New

92 | | 92

[redacted] Why water level allowed to go to -205 for steam cooling

Conditions on Unit 1 are such that EO-100-102, "RPV Control", requires steam cooling.

If the coolant inventory remaining in the reactor vessel is the source of steam for steam cooling, which of the following describes why water level must reach -205 inches before initiating steam cooling?

[redacted] The reduced level ensures the uncovered fuel will be hot enough to provide a large differential temperature between it and the steam being generated allowing maximum heat removal.

[redacted] Core temperatures will lower, allowing additional time for restoration of an injection source before the rapid depressurization is required.

[redacted] Allowing level to lower will reduce the reactor core differential pressure assisting the thermal driving head for natural circulation flow.

[redacted] This level ensures the initial swell upon depressurization will sweep enough coolant past the fuel to break up the boundary layer maximizing heat transfer.

[redacted] a [redacted] S [redacted] Comprehension [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions | RO Group 1 | SRO Group 1

295031 Reactor Low Water Level

EK3. Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL:

EK3.04 Steam cooling | 4.0 | 4.3

[redacted] a. - correct answer, large delta T required for steam cooling to work, need to heat the uncovered fuel so the steam passing by can remove enough heat to minimize fuel damage b. - lowering level will raise fuel temps c. - not a concern or requirement for these conditions d. - should be little to no boundary layer for these conditions

Reference Title	Facility Reference Number	Section	Page Number(s)	Revision	L.O.
RPV Control	EO-000-102	RC/L-22	26	0	
Emergency Operating Procedures	PP002A			4	5 & 7

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 93 [redacted] [redacted] 93

Unit differences on RCIC/HPCI operations in EOP-104/204

While operating in EO-100-104/204, "Secondary Containment Control", the Max Safe Temperatures for the HPCI Equipment Areas are different between Unit 1 (300 degrees F) and Unit 2 (240 degrees F).

Which of the following describes the reason for this difference and how that difference will affect operation in Secondary Containment Control?

- The Unit 2 HPCI Room room coolers are arranged differently and can be provided with cooling from both DX Units. This additional cooling capacity allows lower EO-204 temperature limits.
- The Unit 2 safe shutdown analysis for HPCI equipment operability concerns during loss of off-site power was more restrictive than that done on Unit 1. Thus, EO-204 requires action earlier than EO-104.
- On Unit 2, temperature instrumentation location for RCIC and HPCI is such that the rooms are considered one "area" for EO-204 purposes. Therefore, the more restrictive RCIC Max Safe Temperature is limiting.
- Post loss of off-site power natural ventilation flow has more heat removal capabilities in the Unit 2 Reactor Building as opposed to Unit 1. Additional equipment operability analysis allows a higher temperature in EO-104.

c | S | Memory | Susquehanna | 5/10/99

Emergency and Abnormal Plant Evolutions | 3 | 2

295032 High Secondary Containment Area Temperature

2.2 Equipment Control

2.3 (multi-unit) Knowledge of the design, procedural, and operational differences between units. 3.1 3.3

a. - DX cooling is for Unit 2 Emergency Switchgear Room Coolers only b. - no difference in Unit analysis c. - correct answer, common blowout tunnel has high temp isolation instruments for RCIC and HPCI on Unit 2, Unit 1 is separate allows higher temp on HPCI d. - Reactor Building designs are the same

Secondary Containment Control	EO-000-104	Step SC/T-4, Table 8	13	0	
Emergency Operating Procedures	PP002A			4	7
Reactor Core Isolation Cooling System	SY017 C-5			1	18

Unit 1 EOP Flowcharts with entry conditions removed

New | Question Modification Method:

94 | | 94

[redacted] EO-104 actions to lower off-site release rates

With Unit 1 at power an EO-100-104, "Secondary Containment Control", entry condition has been received.

Which of the following EO-100-104 directed actions will NOT reduce any current and future Off-Site doses for these conditions?

[redacted] "Go to RPV Control" - Step SC/R-5

[redacted] "Rapid Depress is required" - Step SC/R-6

[redacted] "Restart RB HVAC" - Step SC-3

[redacted] "Isolate all systems discharging into area" - Step SC/R-1

[redacted] c [redacted] S [redacted] Comprehension [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 2

295033 High Secondary Containment Area Radiation Levels

EA1. Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS:

EA1.03 Secondary containment ventilation 3.6 3.6

[redacted] a, b, & d. - all reduce the driving head of any leak and/or energy being released c. - RB HVAC provides for monitored, high level release but does NOT treat that release, off-site doses not affected with or without RB HVAC running

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Secondary Containment Control	EO-000-104	Step SC-3	8	0	
Emergency Operating Procedures	PP002A			4	7

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted] [redacted]

[redacted] 95 [redacted] [redacted] 95

[redacted] Determination of Max Safe Water Levels in Secondary Containment

Given the following conditions:

- A confirmed fuel failure has occurred on Unit 1 resulting in a Main Steam Isolation Valve closure
- The HPCI Equipment Area high water level alarm was received just after the Safety Relief Valves opened on the scram
- Suppression pool water level is lowering
- The Reactor Building general area radiation levels are 7.5 rem/hour

Which of the following describes how the water level in the HPCI Equipment Area should be determined in order to take the actions as required in EO-100-104, "Secondary Containment Control"?

[redacted] Assume the water level in the HPCI Equipment Area is above Max Safe Water Level.

[redacted] Calculate the suppression pool water level loss rate and assume it is all going to the HPCI Equipment Area.

[redacted] Obtain a dose extension authorization and attempt a direct observation of HPCI Equipment Area water level.

[redacted] Calculate the Reactor Building Floor Drain Sump Pump run times and extrapolate that value to a HPCI Equipment Area water level.

[redacted] a [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 3 [redacted] 2

95036 Secondary Containment High Sump/Area Water Level

A2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL:

EA2.02 Water level in the affected area 3.1 3.1

[redacted] a. - correct answer, per EO-104 b. - may provide backup to assumption made in a. but not procedurally directed nor may there be time to perform this c. - not required, too time consuming d. - not a viable method for this level determination

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Secondary Containment Control	EO-000-104	SC/L-6	29	0	
Emergency Operating Procedures	PP002A			4	6 & 7

Material Required for Examination: None

[redacted] New Question Modification Method:

[redacted]

[redacted]	[redacted]

Record Number: 96 RO Number: SRO Number: 96

[redacted] Commence injection during an ATWS with Rapid Depress

Given the following conditions:

- Unit 1 had a main turbine trip from 95% power
- 125 control rods did NOT insert on the scram
- High Pressure Coolant Injection is not available
- The Unit Supervisor determined that reactor water level could not be maintained > -161" and directed a Rapid Depressurization
- All injection to the reactor (except CRD, SLC and RCIC) has been stopped and prevented

The Unit Supervisor shall direct restarting injection flow to the reactor when:

- reactor power is less than 5%.
- reactor water level is -205 inches.
- reactor pressure is less than 152 psig.
- 6 ADS Safety Relief Valves have been confirmed open.

[redacted]	d	[redacted]	S	[redacted]	Application	[redacted]	Susquehanna	[redacted]	5/10/99
[redacted]	Emergency and Abnormal Plant Evolutions	[redacted]	1	[redacted]	1				

295037	SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown		
2.1	Conduct of Operations		
2.1.7	Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	3.7	4.4

[redacted] a. - common point at which injection can begin again, previous EOPs b. - 2/3 core coverage, Rapid Depress point without an ATWS c. - pressure to restart injection while in RPV Flooding d. - correct answer, when Rapid Depress initiated

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Level/Power Control	EO-100-113 Sheet 1	Step LQ/L-19		0	
Emergency Operating Procedures	PP002A			4	6

[redacted] Unit 1 EOP Flowcharts with entry conditions removed

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 97 [redacted] [redacted] 97

[redacted] Table 15 systems/LPCI injection during an ATWS

While operating in accordance with EO-100-113, "Level/Power Control", the operator is directed to lower level to between -60 and -110 inches (Step LQ/Q-6) utilizing Table 15 systems.

(

Which of the following describes why the Low Pressure Coolant Injection (LPCI) mode of Residual Heat Removal (RHR) system is the LEAST preferred Table 15 system for accomplishing this step?

[redacted] Utilizing the other Table 15 systems first maintains RHR available for containment and/or suppression pool problems during the ATWS.

[redacted] The LPCI injection flowpath receives minimal preheating and its use may result in power/flow instabilities as level is lowered.

[redacted] The relatively low RHR Pump shutoff head limits the systems' ability to inject during an high power/pressure ATWS.

[redacted] The high RHR Pump flow rates may result in sweeping any injected boron out of the core resulting in a power rise as level is lowered.

[redacted] b [redacted] S [redacted] Comprehension [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 1 [redacted] 1

295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

EK3. Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN:

EK3.03 Lowering reactor water level 4.1 4.5

[redacted] a. - RHR cannot be diverted until adequate core cooling is assured b. - correct answer, LPCI injects into lower plenum, other major systems inject into feedwater sparger or higher outside the shroud, more time for preheating before reaching lower plenum and core entry c. - true conditions but not applicable to this step d. - other systems have as high or higher flow rate, injection would be throttle here anyway

(

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Level/Power Control	EO-000-113	Step LQ/Q-6	19	0	
Emergency Operating Procedures	PP002A			4	7 & 16

[redacted] None

[redacted] New [redacted]

Question Source Comments:

[redacted]

[redacted] 88 [redacted] [redacted] 98

[redacted] Isolation of systems during releases

EO-100-105, "Radioactivity Release Control", directs isolation of all primary systems discharging into areas outside Primary Containment or Reactor Building except those systems required to support EOP/DSP actions.

These systems are specifically exempted from isolation because:

- additional off-site releases from them are unlikely.
- they are required to support alternate reactor depressurization methods.
- their isolation may result in larger, uncontrolled releases as the transient continues.
- these additional isolations would require an unnecessarily escalation of the emergency classification.

[redacted] c [redacted] S [redacted] Memory [redacted] Susquehanna [redacted] 5/10/99

[redacted] Emergency and Abnormal Plant Evolutions [redacted] 2 [redacted] 1

295038 High Off-Site Release Rate

EK3. Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE:

EK3.02 System isolations 3.9 4.2

[redacted] a. - not true b. - alternate depress methods are part of EOPs c. - correct answer d. - not a consideration for these conditions

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
Radioactivity Release Control	EO-000-105	Step RR-2	4	0	
Emergency Operating Procedures	PP002A			4	7

[redacted] None

[redacted] New [redacted]

[redacted]

[redacted]	[redacted]

[redacted] 99 [redacted] [redacted] 99

Actions for a fire with Control Room Evacuation

Unit 1 is operating in accordance with ON-113-001, "Response To Fire".

Select the specific conditions that direct the Unit Supervisor to EXIT ON-113-001 even with a fire still burning.

- The fire is affecting Unit equipment required to reach and maintain "safe shutdown".
- The Fire Brigade Leader has determined that off-site fire fighting assistance is required.
- Any Emergency Operating Procedure entry condition is met.
- ON-100-009, "Control Room Evacuation", entry is required.

d	S	Memory						

Emergency and Abnormal Plant Evolutions	2	2

600000	Plant Fire On Site
EK3.	Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE:
EK3.04	Actions contained in the abnormal procedure for plant fire on site — 2.8 — 3.4

a. - ON-113-001 addresses this concern b. - Still required to operate IAW ON-113-001 c. - ON-113-001 directs EOP entry within 15 minutes d. - correct answer

Response To Fire	ON-113-001	3.5	4	8	
Off-Normal Procedures	AD045			0	3

None

New

Question Source Comments:

100 | | 100

(TAB)

SUSQUEHANNA

NRC WRITTEN EXAM

DATABASE REPORTS

Question Source

Question Source	Question Modification Method	Number
New		83
NRC Exam Bank	Concept Used	3
NRC Exam Bank	Editorially Modified	7
NRC Exam Bank	Significantly Modified	3
Previous 2 NRC Exams	Concept Used	1
Previous 2 NRC Exams	Editorially Modified	2
Previous 2 NRC Exams	Significantly Modified	1

Question Source Comments

Exam Level	KA	Question Source Comments
S	201001K4 K4.04	Grand Gulf NRC Exam (07/95) - cleanup stem and distractors, one new distractor
S	201002K1 K1.01	
S	201003A4 A4.02	
S	201006A2 A2.05	
S	202001A2 A2.08	
S	202002A1 A1.01	Hope Creek NRC Exam 02/98 - modified for SSES specific Recirc numbers, changed ste
S	203000K2 K2.03	
S	204000A3 A3.03	
S	205000K3 K3.02	
S	206000K5 K5.05	
S	206000K6 K6.09	
S	209001K1 K1.01	
S	211000A2 A2.01	
S	211000K3 K3.01	
S	212000A2 A2.19	SSES NRC Exam 09/96 - changed stem to bullet format, used specific valve numbers in
S	212000G 2.1.12	
S	215001K4 K4.01	Hope Creek NRC Exam 02/98 - rewrote stem to reverse the question to have operable sq
S	215002K6 K6.05	
S	215003A4 A4.07	
S	215004G 2.2.26	
S	215005A1 A1.07	
S	216000K3 K3.24	
S	217000A3 A3.01	
S	217000A4 A4.01	
S	218000K2 K2.01	
S	218000K5 K5.01	SSES NRC Exam 04/96 - changed to bullet format, added amplifying conditions
S	223002K4 K4.05	
S	234000K5 K5.02	SSES NRC Exam 04/96 - changed stem to bullet format with additional information, two

Exam Level	KA		Question Source Comments
S	239001K6	K6.02	
S	239002K5	K5.06	VY NRC Exam 01/99 - changed stem to bullet format, modified distractors, new correct
S	241000K6	K6.20	
S	245000A1	A1.07	
S	259002A4	A4.10	SSES NRC Exam 04/96 - simplified the stem and distractors, 2 new distractors, modified
S	261000K1	K1.06	
S	262001A1	A1.05	
S	263000G	2.1.12	SSES NRC Exam 09/96 - changed question from cell voltage to specific gravity question
S	264000A3	A3.01	
S	272000K4	K4.01	
S	286000K3	K3.03	
S	290002A2	A2.05	
S	294001G	2.1.9	
S	294001G	2.1.12	
S	294001G	2.1.12	
S	294001G	2.1.14	
S	294001G	2.1.21	
S	294001G	2.2.13	River Bend NRC Exam (01/97) - changed question from tag removal to checkoff list, one
S	294001G	2.2.14	
S	294001G	2.2.23	
S	294001G	2.3.1	
S	294001G	2.3.2	SSES NRC Exam 08/94 - same concept but different numbers utilized in each distractor
S	294001G	2.3.4	
S	294001G	2.3.10	
S	294001G	2.4.25	
S	294001G	2.4.38	Duane Arnold NRC Exam (07/98) - modified to fit SSES specific titles and positions
S	294001G	2.4.40	
S	294001G	2.4.41	
S	294001G	2.4.49	
S	295001A1	AA1.01	

Exam Level	KA	Question Source Comments
S	295001K1 AK1.02	River Bend NRC Exam 01/97 - changed stem to bullet format and SSES specific data, on
S	295002K2 AK2.05	
S	295003A1 AA1.03	
S	295003K2 AK2.04	
S	295004A1 AA1.02	
S	295005G 2.4.48	
S	295006K1 AK1.02	
S	295007A1 AA1.04	
S	295009G 2.4.12	
S	295009K1 AK1.05	
S	295010K1 AK1.01	
S	295010K2 AK2.02	
S	295012K2 AK2.02	
S	295013A1 AA1.02	
S	295013K3 AK3.02	
S	295014K2 AK2.02	
S	295015A2 AA2.01	
S	295016A2 AA2.03	
S	295016K2 AK2.01	
S	295017K1 AK1.02	
S	295018K2 AK2.02	
S	295019A2 AA2.02	
S	295020K3 AK3.04	
S	295021A1 AA1.04	
S	295022K1 AK1.01	
S	295023K2 AK2.02	
S	295024A2 EA2.04	Peach Bottom NRC Exam 09/97 - different stem conditions, one new distractor
S	295025K1 EK1.03	River Bend NRC Exam 01/97 - modified stem and distractors to SSES SRV Safety setpoi
S	295026A2 EA2.01	Hope Creek NRC Exam 08/94 - used idea, modified to SSES Improved Tech Spec numb
S	295026G 2.4.1	

Exam Level	KA	Question Source Comments
S	295028K1 EK1.01	Peach Bottom NRC Exam 09/98 - cleaned up stem, made question ECCS system specific
S	295029A2 EA2.03	
S	295030A2 EA2.01	
S	295030K3 EK3.03	
S	295031K3 EK3.04	
S	295032G 2.2.3	
S	295033A1 EA1.03	
S	295036A2 EA2.02	
S	295037G 2.1.7	
S	295037K3 EK3.03	
S	295038K3 EK3.02	
S	600000K3 EK3.04	

Material Required for Examination Administration

Exam Level	*KA*	*MaterialRequiredforExamination*
S	201001 K4.04	None
	201002 K1.01	None
	201003 A4.02	None
	201006 A2.05	None
	202001 A2.08	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	202002 A1.01	None
	203000 K2.03	None
	204000 A3.03	None
	205000 K3.02	None
	206000 K5.05	None
	206000 K6.09	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	209001 K1.01	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	211000 A2.01	None
	211000 K3.01	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	212000 A2.19	None
	212000 2.1.12	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	215001 K4.01	Valve Control Monitor Panel, Figure 14 SY017 I-5
	215002 K6.05	None
	215003 A4.07	None
	215004 2.2.26	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	215005 A1.07	None
	216000 K3.24	None
	217000 A3.01	None
	217000 A4.01	None
	218000 K2.01	None
	218000 K5.01	None
	223002 K4.05	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	234000 K5.02	None
	239001 K6.02	None
	239002 K5.06	None
	241000 K6.20	Figure 8 from SY017 A-8 EHC Logic Diagram
	245000 A1.07	None
	259002 A4.10	None
	261000 K1.06	None

Exam Level	KA	MaterialRequiredforExamination
S	262001 A1.05	None
	263000 2.1.12	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	264000 A3.01	None
	272000 K4.01	None
	286000 K3.03	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	290002 A2.05	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	GENERIC 2.1.9	None
	GENERIC 2.1.12	None
	GENERIC 2.1.12	None
	GENERIC 2.1.14	None
	GENERIC 2.1.21	None
	GENERIC 2.2.13	None
	GENERIC 2.2.14	None
	GENERIC 2.2.23	None
	GENERIC 2.3.1	None
	GENERIC 2.3.2	None
	GENERIC 2.3.4	None
	GENERIC 2.3.10	None
	GENERIC 2.4.25	None
	GENERIC 2.4.38	None
	GENERIC 2.4.40	None
	GENERIC 2.4.41	None
	GENERIC 2.4.49	None
	295001 AA1.01	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	295001 AK1.02	None
	295002 AK2.05	None
	295003 AA1.03	None
	295003 AK2.04	None
	295004 AA1.02	None
	295005 2.4.48	None
	295006 AK1.02	None
	295007 AA1.04	Unit 1 EOP Flowcharts with entry conditions removed
	295009 2.4.12	Figure 1 from SY017 L-1
	295009 AK1.05	None
	295010 AK1.01	None
	295010 AK2.02	None

Exam Level	KA	MaterialRequiredforExamination
S	295012 AK2.02	None
	295013 AA1.02	None
	295013 AK3.02	None
	295014 AK2.02	None
	295015 AA2.01	None
	295016 AA2.03	None
	295016 AK2.01	None
	295017 AK1.02	None
	295018 AK2.02	None
	295019 AA2.02	None
	295020 AK3.04	None
	295021 AA1.04	None
	295022 AK1.01	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	295023 AK2.02	None
	295024 EA2.04	None
	295025 EK1.03	Steam Tables - Mollier Diagram
	295026 EA2.01	Unit 1 Tech Specs Index and Sections 3.1 thru 3.10, w/o bases
	295026 2.4.1	None
	295028 EK1.01	None
	295029 EA2.03	None
	295030 EA2.01	Unit 1 EOP Flowcharts with entry conditions removed
	295030 EK3.03	None
	295031 EK3.04	None
	295032 2.2.3	Unit 1 EOP Flowcharts with entry conditions removed
	295033 EA1.03	None
	295036 EA2.02	None
	295037 2.1.7	Unit 1 EOP Flowcharts with entry conditions removed
	295037 EK3.03	None
	295038 EK3.02	None
	600000 EK3.04	None

Question Cross Reference

KA	Record Number	Exam Level	RO	SRO
GENERIC 2.1.9	1	S		1
GENERIC 2.1.12	2	S		2
GENERIC 2.1.12	3	S		3
GENERIC 2.1.14	4	S		4
GENERIC 2.1.21	5	S		5
GENERIC 2.2.13	6	S		6
GENERIC 2.2.14	7	S		7
GENERIC 2.2.23	8	S		8
GENERIC 2.3.1	9	S		9
GENERIC 2.3.2	10	S		10
GENERIC 2.3.4	11	S		11
GENERIC 2.3.10	12	S		12
GENERIC 2.4.25	13	S		13
GENERIC 2.4.38	14	S		14
GENERIC 2.4.40	15	S		15
GENERIC 2.4.41	16	S		16
GENERIC 2.4.49	17	S		17
201001 K4.04	18	S		18
201002 K1.01	19	S		19
201003 A4.02	20	S		20
201006 A2.05	21	S		21
202001 A2.08	22	S		22
202002 A1.01	23	S		23
203000 K2.03	24	S		24
204000 A3.03	25	S		25
205000 K3.02	26	S		26
206000 K5.05	27	S		27
206000 K6.09	28	S		28
209001 K1.01	29	S		29
211000 A2.01	30	S		30
211000 K3.01	31	S		31
212000 A2.19	32	S		32
212000 2.1.12	33	S		33
215001 K4.01	34	S		34
215002 K5.05	35	S		35
215003 A4.07	36	S		36
215004 2.2.26	37	S		37
215005 A1.07	38	S		38
216000 K3.24	39	S		39
217000 A3.01	40	S		40
217000 A4.01	41	S		41

KA	Record Number	Exam Level	RO	SRO
218000 K2.01	42	S		42
218000 K5.01	43	S		43
223002 K4.05	44	S		44
234000 K5.02	45	S		45
239001 K6.02	46	S		46
239002 K5.06	47	S		47
241000 K6.20	48	S		48
245000 A1.07	49	S		49
259002 A4.10	50	S		50
261000 K1.06	51	S		51
262001 A1.05	52	S		52
263000 2.1.12	53	S		53
264000 A3.01	54	S		54
272000 K4.01	55	S		55
286000 K3.03	56	S		56
290002 A2.05	57	S		57
295001 AA1.01	58	S		58
295001 AK1.02	59	S		59
295002 AK2.05	60	S		60
295003 AA1.03	61	S		61
295003 AK2.04	62	S		62
295004 AA1.02	63	S		63
295005 2.4.48	64	S		64
295006 AK1.02	65	S		65
295007 AA1.04	66	S		66
295009 2.4.12	67	S		67
295009 AK1.05	68	S		68
295010 AK1.01	69	S		69
295010 AK2.02	70	S		70
295012 AK2.02	71	S		71
295013 AA1.02	72	S		72
295013 AK3.02	73	S		73
295014 AK2.02	74	S		74
295015 AA2.01	75	S		75
295016 AA2.03	76	S		76
295016 AK2.01	77	S		77
295017 AK1.02	78	S		78
295018 AK2.02	79	S		79
295019 AA2.02	80	S		80
295020 AK3.04	81	S		81
295021 AA1.04	82	S		82
295022 AK1.01	83	S		83

KA	Record Number	Exam Level	RO	SRO
295023 AK2.02	84	S		84
295024 EA2.04	85	S		85
295025 EK1.03	86	S		86
295026 EA2.01	87	S		87
295026 2.4.1	88	S		88
295028 EK1.01	89	S		89
295029 EA2.03	90	S		90
295030 EA2.01	91	S		91
295030 EK3.03	92	S		92
295031 EK3.04	93	S		93
295032 2.2.3	94	S		94
295033 EA1.03	95	S		95
295036 EA2.02	96	S		96
295037 2.1.7	97	S		97
295037 EK3.03	98	S		98
295038 EK3.02	99	S		99
600000 EK3.04	100	S		100

Exam Level Count

Exam Level	Total Of KA
S	100

SRO Answer Distribution

Answer	*Number of Questions*
a	25
b	25
c	25
d	25

SRO Cognitive Level

Cognitive Level	Number of Questions
Application	32
Comprehension	27
Memory	41