

**INITIAL SUBMITTAL OF THE WRITTEN EXAMINATION**

**FOR THE PERRY EXAMINATION ON JUNE 19, 2001**

May 16, 2001  
PY-CEI/OIE-0540L

United States Nuclear Regulatory Commission  
801 Warrenville Road  
Lisle, Illinois 60532-4351

Attention: Mr. Michael Bielby, Chief Examiner  
Operations Branch Region III

Perry Nuclear Power Plant  
Docket No. 50-440  
NRC Initial License Written Examination

Dear Mr. Bielby:

In accordance with NUREG-1021, ES-201, enclosed are the proposed written examinations, supporting documentation, and reference materials for the NRC initial license written examination tentatively scheduled to be administered on June 19, 2001. The written examination materials were developed in accordance with the guidelines specified in NUREG-1021, Revision 8, Supplement 1 (draft).

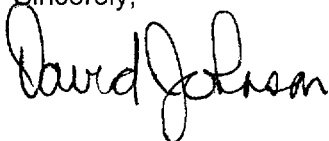
NUREG-1021, Form ES-401-7, is enclosed. In addition, written exam questions are on facility-generated forms with the same documentation as contained on Form ES-401-6.

The NRC initial license written examination outlines were previously submitted with a cover letter dated March 1, 2001. On March 7, 2001, the results of your written examination outline review were discussed with the Facility Author. No corrections or changes were made to the original written examination outlines. In addition, under separate cover letters dated March 29, 2001, and April 18, 2001, revisions were submitted and subsequently approved for the reactor operator and senior reactor operator written examination outlines respectively.

Pursuant to NUREG 1021, ES-201, the enclosed examination materials shall be withheld from public disclosure until after the initial license written examinations are completed. To simplify your review, enclosed in the blue folder, Tab 1, is a packing list that references the specific folder, tab location, and the type of examination material.

If you require any other clarification concerning these items, please call me at (440) 280-5277 or (440) 280-5558.

Sincerely,



David Johnson  
Nuclear Qualification Instructor



James Kelly  
Facility Reviewer

MAY 18 2001

---

Submitted  
Exam (before  
comments)

---

Facility: <u>PERRY</u>		Date of Exam: <u>June 2001</u>		Exam Level: <u>RO/SRO</u>		
Item Description				Initial		
				a	b*	c*
1.	Questions and answers technically accurate and applicable to facility			DPJ	JK	
2.	a. NRC K/As referenced for all questions b. Facility learning objectives referenced as available			DPJ	JK	
3.	RO/SRO overlap is no more than 75 percent, and SRO questions are appropriate per Section D.2.d of ES-401			DPJ	JK	
4.	Question selection and duplication from the last two NRC licensing exams appears consistent with a systematic sampling process					
5.	Question duplication from the license screening/audit exam was controlled as indicated below (check the item that applies) and appears appropriate: <input type="checkbox"/> the audit exam was systematically and randomly developed; or <input checked="" type="checkbox"/> the audit exam was completed before the license exam was started; or <input checked="" type="checkbox"/> the examinations were developed independently; or <input type="checkbox"/> the licensee certifies that there is no duplication; or <input type="checkbox"/> other (explain)			DPJ	JK	
6.	Bank use meets limits (no more than 75 percent from the bank at least 10 percent new, and the rest modified); enter the actual question distribution at right	Bank	Modified	New	DPJ	JK
		27	7	66		
7.	Between 50 and 60 percent of the questions on the exam (including 10 new questions) are written at the comprehension/analysis level; enter the actual question distribution at right	Memory	C/A		DPJ	JK
		48	52			
8.	References/handouts provided do not give away answers			DPJ	JK	
9.	Question content conforms with specific K/A statements in the previously approved examination outline and is appropriate for the Tier to which they are assigned; deviations are justified			DPJ	JK	
10.	Question psychometric quality and format meet ES, Appendix B, guidelines			DPJ	JK	
11.	The exam contains 100, one-point, multiple choice items; the total is correct and agrees with value on cover sheet			DPJ	JK	
a. Author		Printed Name / Signature <u>David P. Johnson</u> <u>David P. Johnson</u>			Date <u>5-15-01</u>	
b. Facility Reviewer (*)		<u>James B. Kelly</u> <u>James B. Kelly</u>			<u>5-15-01</u>	
c. NRC Chief Examiner (#)						
d. NRC Regional Supervisor						
Note: * The facility reviewer's initials/signature are not applicable for NRC-developed examinations. # Independent NRC reviewer initial items in Column "c;" chief examiner concurrence required.						

The licensee  
count is wrong.  
It should be 53 @  
higher level.  
JH

Facility: <u>PERRY</u>		Date of Exam: <u>JUNE 2001</u>		Exam Level: <u>RO/SRO</u>	
Item Description	Initial				
	a	b*	c*		
1. Questions and answers technically accurate and applicable to facility	DPJ	JK			
2. a. NRC K/As referenced for all questions b. Facility learning objectives referenced as available	DPJ	JK			
3. RO/SRO overlap is no more than 75 percent, and SRO questions are appropriate per Section D.2.d of ES-401	DPJ	JK			
4. Question selection and duplication from the last two NRC licensing exams appears consistent with a systematic sampling process					
5. Question duplication from the license screening/audit exam was controlled as indicated below (check the item that applies) and appears appropriate: <input type="checkbox"/> the audit exam was systematically and randomly developed; or <input checked="" type="checkbox"/> the audit exam was completed before the license exam was started; or <input type="checkbox"/> the examinations were developed independently; or <input type="checkbox"/> the licensee certifies that there is no duplication; or <input type="checkbox"/> other (explain)	DPJ	JK			
6. Bank use meets limits (no more than 75 percent from the bank at least 10 percent new, and the rest modified); enter the actual question distribution at right	Bank 23	Modified 8	New 69	DPJ	JK
7. Between 50 and 60 percent of the questions on the exam (including 10 new questions) are written at the comprehension/analysis level; enter the actual question distribution at right	Memory 49	C/A 51		DPJ	JK
8. References/handouts provided do not give away answers	DPJ	JK			
9. Question content conforms with specific K/A statements in the previously approved examination outline and is appropriate for the Tier to which they are assigned; deviations are justified	DPJ	JK			
10. Question psychometric quality and format meet ES, Appendix B, guidelines	DPJ	JK			
11. The exam contains 100, one-point, multiple choice items; the total is correct and agrees with value on cover sheet	DPJ	JK			
a. Author b. Facility Reviewer (*) c. NRC Chief Examiner (#) d. NRC Regional Supervisor		Printed Name / Signature <u>David P. Johnson</u> <u>David P. Johnson</u> <u>James B. Kelly</u> <u>James B. Kelly</u>		Date <u>5-15-01</u> <u>5-15-01</u>	
Note: * The facility reviewer's initials/signature are not applicable for NRC-developed examinations. # Independent NRC reviewer initial items in Column "c;" chief examiner concurrence required.					

Facility: Perry		Date of Exam: June 2001				Exam Level: RO							
Tier	Group	K/A Category Points											Point Total
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	
1. Emergency & Abnormal Plant Evolutions	1	1	3	2				3	4			0	13
	2	3	1	6				5	2			2	19
	3	1	1	1				0	1			0	4
	Tier Totals	5	5	9				8	7			2	36
2. Plant Systems	1	2	2	2	3	0	2	4	5	5	2	1	28
	2	4	0	2	2	2	0	0	3	3	2	1	19
	3	1	0	0	0	1	0	0	1	1	0	0	4
	Tier Totals	7	2	4	5	3	2	4	9	9	4	2	51
3. Generic Knowledge and Abilities					Cat 1		Cat 2		Cat 3		Cat 4		13
					3		3		2		5		
<p>Note: 1. Ensure that at least two topics from every K/A category are sampled within each tier (i.e., the "Tier Totals" in each K/A category shall not be less than two).</p> <p>2. Actual point totals must match those specified in the table.</p> <p>3. Select topics from many systems; avoid selecting more than two or three K/A topics from a given system unless they relate to plant-specific priorities.</p> <p>4. Systems/evolutions within each group are identified on the associated outline.</p> <p>5. The shaded areas are not applicable to the category/tier.</p> <p>6.* The generic K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.</p> <p>7. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings for the RO license level, and the point totals for each system and category. K/As below 2.5 should be justified on the basis of plant-specific priorities. Enter the tier totals for each category in the table above.</p>													

ES-401		BWR RO Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 1						Form ES-401-2	
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295005 Main Turbine Generator Trip / 3		3					Recirculation System	3.2	1
295006 SCRAM / 1		2					Reactor Water Level Control System	3.8	1
295007 High Reactor Pressure / 3				5			Reactor/Turbine Pressure Regulating System	3.7	1
295009 Low Reactor Water Level / 2					1		Reactor Water Level	4.2	1
295010 High Drywell Pressure / 5				2			Drywell Floor And Equipment Drain Sumps	3.6	1
295014 Inadvertent Reactivity Addition / 1					1		Reactor Power	4.1	1
295015 Incomplete SCRAM / 1	4	8					Reactor Pressure	3.8	1
							Neutron Monitoring System	3.6	1
295024 High Drywell Pressure / 5					1		Drywell Pressure	4.2	1
295025 High Reactor Pressure / 3					3		Suppression Pool Temperature	3.9	1
295031 Reactor Low Water Level / 2				12			Feedwater	3.9	1
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1			5				Cold Shutdown Boron Weight	3.2	1
500000 High Containment Hydrogen Conc. / 5			3				Operation of Hydrogen Recombiners	3.0	1
K/A Category Totals:	1	3	2	3	4	0	Group Point Total:		13

ES-401		BWR RO Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 2						Form ES-401-2	
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4						2.4.11	Knowledge Of Abnormal Condition Procedures	3.4	1
295002 Loss of Main Condenser Vacuum / 3				5			Main Turbine	3.2	1
295003 Partial or Complete Loss of AC Pwr / 6	2						Load Shedding	3.1	1
295004 Partial or Complete Loss of DC Pwr / 6			3				Reactor SCRAM	3.1	1
295008 High Reactor Water Level / 2	2						Component Erosion/Damage	2.8	1
295011 High CTMT Temperature / 5			1				Increased Containment Cooling	3.6	1
295012 High Drywell Temperature / 5					2		Drywell Pressure	3.9	1
295013 High Suppression Pool Temp. / 5						2.1.12	Ability To Apply Technical Specifications For A System	2.9	1
295016 Control Room Abandonment / 7				4			AC Electrical Distribution	3.1	1
295017 High Off-site Release Rate / 9				3			Plant Ventilation Systems	3.4	1
295018 Partial or Complete Loss of CCW / 8			6				Increasing Cooling Water Flow To Heat Exchangers	3.3	1
295019 Part. or Comp. Loss of Inst. Air / 8									
295020 Inadvertent Cont. Isolation / 5 & 7									
295022 Loss of CRD Pumps / 1		3					Accumulator Pressures	3.4	1
295026 High Suppression Pool Water Temp. / 5					1		Suppression Pool Water Temperature	4.1	1
295027 High Containment Temperature / 5			3				Reactor SCRAM	3.7	1
295028 High Drywell Temperature / 5									
295029 High Suppression Pool Water Level / 5	1						Containment Integrity	3.4	1
295030 Low Suppression Pool Water Level / 5			3				RCIC Operation	3.6	1
295033 High Sec. Cont. Area Rad. Levels / 9				1			Area Radiation Monitoring System	3.9	1
295034 Sec. Cont. Ventilation High Rad. / 9				3			Secondary Containment Ventilation	4.0	1
295038 High Off-site Release Rate / 9			2				System Isolations	3.9	1
600000 Plant Fire On Site / 8									
K/A Category Point Totals:	3	1	6	5	2	2	Group Point Total:		19



ES-401		BWR RO Examination Outline Emergency and Abnormal Plant Evolutions - Tier 1/Group 3						Form ES-401-2	
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points
295021 Loss of Shutdown Cooling / 4	1						Decay Heat	3.6	1
295023 Refueling Accidents / 8					2		Fuel Pool Level	3.4	1
295032 High Secondary Containment Area Temperature / 5			3				Isolating Affected Systems	3.8	1
295035 Secondary Containment High Differential Pressure / 5									
295036 Secondary Containment High Sump/Area Water Level / 5		1					Secondary Containment Equipment And Floor Drain System	3.1	1
K/A Category Point Totals:	1	1	1	0	1	0	Group Point Total:		4

BWR RO Examination Outline Plant Systems - Tier 2/Group 1													Form ES-401-2	
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201001 CRD Hydraulic							2					CRD Cooling Water Header Pressure	2.9	1
201002 RMCS														
201005 RCIS				6						1		Rod Pattern Controller Rod Blocks Operator Control Module (Lights & Push Buttons)	3.5 3.7	1 1
202002 Recirculation Flow Control								6	1			Low Reactor Water Level Flow Control Valve Operation	3.3 3.6	1 1
203000 RHR/LPCI: Injection Mode	10											ECCS Room Coolers	3.2	1
206000 HPCI														
207000 Isolation (Emerg.) Condenser														
209001 LPCS	14						4					Reactor Vessel Reactor Pressure	3.7 3.7	1 1
209002 HPCS		3							2			Initiation Logic Pump Start	2.8 3.8	1 1
211000 SLC							1					Tank Level	3.6	1
212000 RPS									5			SCRAM Instrument Volume Level	3.9	1
215003 IRM								5				Faulty Or Erratic Operation Of Detectors/System	3.3	1
215004 SRM				1								Rod Withdrawal Blocks	3.7	1
215005 APRM / LPRM										1		IRM/APRM Recorder	3.2	1
216000 Nuclear Boiler Instrumentation								6				Loss Of Power Supply	2.9	1
217000 RCIC						1						Electrical Power	3.4	1
218000 ADS		1										ADS Logic	3.1	1
223001 Primary CTMT and Auxiliaries						14		4				RHR/LPCI High Cntmt/Drywell Hydrogen Concentration	3.6 3.7	1 1
223002 PCIS/Nuclear Steam Supply Shutoff									2			Valve Closures	3.5	1
239002 SRVs								1				Stuck Open Vacuum Breakers	3.0	1
241000 Reactor/Turbine Pressure Regulator			17									Turbine Acceleration	2.7	1
259001 Reactor Feedwater											2.4.6	Knowledge Symptom Based EOP Mitigation Strategies	3.1	1
259002 Reactor Water Level Control				8			1					TDRFP Speed Control Reactor Water Level	2.9 3.8	1 1
261000 SGTS									3			Valve Operation	3.0	1
264000 EDGs			2									AC Electrical Distribution	3.9	1
K/A Category Point Totals:	2	2	2	3	0	2	4	5	5	2	1	Group Point Total:		28

BWR RO Examination Outline Plant Systems - Tier 2/Group 2														Form ES-401-2
ES-401														
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201003 Control Rod and Drive Mechanism			1									Reactor Power	3.2	1
201004 RSCS														
201006 RWM														
202001 Recirculation										9		Reactor Water Level	3.7	1
204000 RWCU	11											PCIS/NSSSS	3.5	1
205000 Shutdown Cooling											2.1.32	Ability To Explain And Apply System Limits And Precautions	3.4	1
214000 RPIS														
215002 RBM														
219000 RHR/LPCI: Torus/Pool Cooling Mode				8								Adequate Pump Net Positive Suction Head	2.9	1
226001 RHR/LPCI: CTMT Spray Mode									7			Pump Start	3.5	1
230000 RHR/LPCI: Torus/Pool Spray Mode														
239001 Main and Reheat Steam					5							Flow Indication	2.8	1
245000 Main Turbine Gen. and Auxiliaries										14		Generator Megavar Output	2.5	1
256000 Reactor Condensate	18											Circulating Water System	2.9	1
262001 AC Electrical Distribution								1				Turbine/Generator Trip	3.4	1
262002 UPS (AC/DC)									1			Transfer From Preferred To Alternate Source	2.8	1
263000 DC Electrical Distribution			2									Components Using DC Control Power (i.e. Breakers)	3.5	1
271000 Offgas	7											Plant Air Systems	2.7	1
272000 Radiation Monitoring	2											Off-Gas System	3.2	1
286000 Fire Protection									1			Fire Water Pump Start	3.4	1
290001 Secondary CTMT														
290003 Control Room HVAC					3							Temperature Control	2.6	1
300000 Instrument Air								1				Air Dryer And Filter Malfunctions	2.9	1
400000 Component Cooling Water				1				1				Automatic Start Of Standby Pump	3.4	1
												Loss Of CCW Pump	3.3	1
K/A Category Point Totals:	4	0	2	2	2	0	0	3	3	2	1	Group Point Total:		19

ES-401		BWR RO Examination Outline Plant Systems - Tier 2/Group 3											Form ES-401-2	
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
215001 Traversing In-core Probe														
233000 Fuel Pool Cooling and Cleanup					1							Heat Removal Mechanisms	2.5	1
234000 Fuel Handling Equipment	6											RC & IS	3.0	1
239003 MSIV Leakage Control														
268000 Radwaste														
288000 Plant Ventilation									1			Isolation/Initiation Signals	3.8	1
290002 Reactor Vessel Internals								5				Exceeding Thermal Limits	3.7	1
K/A Category Point Totals:	1	0	0	0	1	0	0	1	1	0	0	Group Point Total:		4
Plant-Specific Priorities														
System / Topic	Recommended Replacement for...					Reason					Points			
295001 / G2.4.11	295001 AK2.02					ONI-C51 Revision (Post RF08)					1			
Plant-Specific Priority Total: (limit 10)											1			

Facility: Perry		Date of Exam: July 2001		Exam Level: RO	
Category	K/A #	Topic	Imp.	Points	
Conduct of Operations	2.1.29	Knowledge Of How To Conduct And Verify Valve Lineups	3.4	1	
	2.1.3	Knowledge Of Shift Turnover Practices	3.0	1	
	2.1.20	Ability To Execute Procedural Steps	4.3	1	
	Total			3	
Equipment Control	2.2.27	Knowledge Of The Refueling Process	2.6	1	
	2.2.11	Knowledge Of The Process For Controlling Temporary Changes	2.5	1	
	2.2.13	Knowledge Of Tagging And Clearance Procedures	3.6	1	
	Total			3	
Radiation Control	2.3.2	Knowledge Of Facility ALARA Program	2.5	1	
	2.3.4	Knowledge Of Radiation Exposure Limits And Contamination Control / Including Permissible Levels In Excess Of Those Authorized	2.5	1	
	Total			2	
Emergency Procedures/ Plan	2.4.1	Knowledge Of EOP Entry Conditions And Immediate Action Steps	4.3	1	
	2.4.5	Knowledge Of The Organization Of The Operating Procedures Network For Normal/Abnormal/And Emergency Evolutions	2.9	1	
	2.4.12	Knowledge Of General Operating Crew Responsibilities During Emergency Operations	3.4	1	
	2.4.9	Knowledge Of Low Power/Shutdown Implications In Accident (E.G. LOCA Or Loss Of RHR) Mitigation Strategies	3.3	1	
	2.4.25	Knowledge Of Fire Protection Procedures	2.9	1	
	Total			5	
Tier 3 Point Total (RO/SRO)				13/17	

Facility: Perry      Date of Exam: July 2001      Exam Level: SRO													
Tier	Group	K/A Category Points											Point Total
		K 1	K 2	K 3	K 4	K 5	K 6	A 1	A 2	A 3	A 4	G *	
1. Emergency & Abnormal Plant Evolutions	1	2	3	5				6	8			2	26
	2	3	3	4				3	1			3	17
	Tier Totals	5	6	9				9	9			5	43
2. Plant Systems	1	1	2	2	2	0	2	2	5	4	1	2	23
	2	2	0	1	2	1	0	0	1	2	2	2	13
	3	0	0	1	0	1	0	0	1	1	0	0	4
	Tier Totals	3	2	4	4	2	2	2	7	7	3	4	40
3. Generic Knowledge and Abilities							Cat 1	Cat 2	Cat 3	Cat 4	17		
							5	4	2	6			
<p>Note: 1. Ensure that at least two topics from every K/A category are sampled within each tier (i.e., the "Tier Totals" in each K/A category shall not be less than two).</p> <p>2. Actual point totals must match those specified in the table.</p> <p>3. Select topics from many systems; avoid selecting more than two or three K/A topics from a given system unless they relate to plant-specific priorities.</p> <p>4. Systems/evolutions within each group are identified on the associated outline.</p> <p>5. The shaded areas are not applicable to the category/tier.</p> <p>6.* The generic K/As in Tiers 1 and 2 shall be selected from Section 2 of the K/A Catalog, but the topics must be relevant to the applicable evolution or system.</p> <p>7. On the following pages, enter the K/A numbers, a brief description of each topic, the topics' importance ratings for the SRO license level, and the point totals for each system and category. K/As below 2.5 should be justified on the basis of plant-specific priorities. Enter the tier totals for each category in the table above.</p>													

ES-401 Emergency and Abnormal Plant Evolutions - Tier 1/Group 1							BWR SRO Examination Outline		Form ES-401-1	
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points	
295003 Partial or Complete Loss of AC Pwr / 6	2						Load Shedding	3.4	1	
295006 SCRAM / 1		2		4			Reactor Water Level Control System	3.8	1	
							Recirculation System	3.2	1	
295007 High Reactor Pressure / 3				5			Reactor/Turbine Pressure Regulating System	3.8	1	
295009 Low Reactor Water Level / 2					1		Reactor Water Level	4.2	1	
295010 High Drywell Pressure / 5				2			Drywell Floor And Equipment Drain Sumps	3.6	1	
295013 High Suppression Pool Temp. / 5						2.1.12	Ability To Apply Technical Specifications For A System	4.0	1	
295014 Inadvertent Reactivity Addition / 1					1		Reactor Power	4.2	1	
295015 Incomplete SCRAM / 1	4	8					Reactor Pressure	3.8	1	
							Neutron Monitoring System	3.7	1	
295016 Control Room Abandonment / 7				4	2		AC Electrical Distribution	3.2	1	
							Reactor Water Level	4.3	1	
295017 High Off-site Release Rate / 9				3		2.4.41	Plant Ventilation Systems	3.4	1	
							Knowledge Of The Emergency Action Level Thresholds & Classifications	4.1	1	
295023 Refueling Accidents Cooling Mode / 8		5			2		Fuel Pool Level	3.7	1	
							RPS	4.0	1	
295024 High Drywell Pressure / 5					1		Drywell Pressure	4.4	1	
295025 High Reactor Pressure / 3					3		Suppression Pool Temperature	4.1	1	
295026 Suppression Pool High Water Temp. / 5					1		Suppression Pool Water Temperature	4.2	1	
295027 High Containment Temperature / 5			3				Reactor SCRAM	3.7	1	
295030 Low Suppression Pool Water Level / 5			3		3		RCIC Operation	3.7	1	
							Reactor Pressure	3.9	1	
295031 Reactor Low Water Level / 2				12			Feedwater	4.1	1	
295037 SCRAM Condition Present and Power Above APRM Downscale or Unknown / 1			5				Cold Shutdown Boron Weight	3.7	1	
295038 High Off-site Release Rate / 9			2				System Isolations	4.2	1	
500000 High Containment Hydrogen Conc. / 5			3				Operation of Hydrogen Recombiners	3.5	1	
K/A Category Totals:	2	3	5	6	8	2	Group Point Total:		26	

ES-401 Emergency and Abnormal Plant Evolutions - Tier 1/Group 2							BWR SRO Examination Outline		Form ES-401-1	
E/APE # / Name / Safety Function	K1	K2	K3	A1	A2	G	K/A Topic(s)	Imp.	Points	
295001 Partial or Complete Loss of Forced Core Flow Circulation / 1 & 4						2.4.11	Knowledge of Abnormal Condition Procedures	3.6	1	
295002 Loss of Main Condenser Vacuum / 3	1						Plant Efficiency	2.5	1	
295004 Partial or Total Loss of DC Pwr / 6			3				Reactor SCRAM	3.5	1	
295005 Main Turbine Generator Trip / 3		3					Recirculation System	3.3	1	
295008 High Reactor Water Level / 2	2						Component/Erosion Damage	2.8	1	
295011 High Containment Temperature / 5			1				Increased Containment Cooling	3.9	1	
295012 High Drywell Temperature / 5					2		Drywell Pressure	4.1	1	
295018 Partial or Total Loss of CCW / 8			6				Increased Cooling Water Flow to Heat Exchangers	3.3	1	
295019 Partial or Total Loss of Inst. Air / 8						2.1.33	Ability to Recognize Indications for System Operating Parameters Which are Entry-Level Conditions for Technical Specifications	4.0	1	
295020 Inadvertent Cont. Isolation / 5 & 7				2			Drywell Ventilation/Cooling System	3.2	1	
295021 Loss of Shutdown Cooling / 4						2.1.25	Ability to Obtain and Interpret Station Reference Materials Such as Graphs/Monographs/ and Tables Which Contain Performance Data	3.1	1	
295022 Loss of CRD Pumps / 1		3					Accumulator Pressures	3.4	1	
295028 High Drywell Temperature / 5										
295029 High Suppression Pool Water Level / 5	1						Containment Integrity	3.7	1	
295032 High Secondary Containment Area Temperature / 5			3				Isolating Affected Systems	3.9	1	
295033 High Secondary Containment Area Radiation Levels / 9				1			Area Radiation Monitoring System	4.0	1	
295034 Secondary Containment Ventilation High Radiation / 9				3			Secondary Containment Ventilation	3.9	1	
295035 Secondary Containment High Differential Pressure / 5										
295036 Secondary Containment High Sump/Area Water Level / 5		1					Secondary Containment Equipment and Floor Drain System	3.2	1	
600000 Plant Fire On Site / 8										
K/A Category Point Totals:	3	3	4	3	1	3	Group Point Total:		17	



ES-401		BWR SRO Examination Outline Plant Systems - Tier 2/Group 1											Form ES-401-1	
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201005 RCIS				6								Rod Pattern Controller Rod Blocks	3.5	1
202002 Recirculation Flow Control									1			Flow Control Valve Operation	3.4	1
203000 RHR/LPCI: Injection Mode	10											ECCS Room Coolers	3.2	1
206000 HPCI														
207000 Isolation (Emergency) Condenser														
209001 LPCS							4					Reactor Pressure	3.7	1
209002 HPCS		3										Initiation Logic	2.9	1
211000 SLC											2.2.24	Ability To Analyze The Affect Of Maintenance Activities On LCO Status	3.8	1
212000 RPS									5			SCRAM Instrument Volume Level	3.9	1
215004 Source Range Monitor								6				Failed Recorder	2.5	1
215005 APRM / LPRM										1		IRM/APRM Recorder	3.1	1
216000 Nuclear Boiler Instrumentation								6				Loss Of Power Supply	3.1	1
217000 RCIC						1						Electrical Power	3.5	1
218000 ADS		1										ADS Logic	3.3	1
223001 Primary CTMT and Auxiliaries						14		4				RHR/LPCI High Cntmt/Drywell Hydrogen Concentration	3.8 3.8	1 1
223002 PCIS/Nuclear Steam Supply Shutoff									2			Valve Closures	3.5	1
226001 RHR/LPCI: CTMT Spray Mode									7			Pump Start	3.5	1
239002 SRVs								1				Stuck Open Vacuum Breakers	3.3	1
241000 Reactor/Turbine Pressure Regulator			17									Turbine Acceleration	2.8	1
259002 Reactor Water Level Control				8			1					TDRFP Speed Control Reactor Water Level	3.0 3.8	1 1
261000 SGTS											2.2.25	Knowledge Of Bases In Technical Specifications For LCOs And Safety Limits	3.7	1
262001 AC Electrical Distribution								1				Turbine/Generator Trip	3.6	1
264000 EDGs			2									AC Electrical Distribution	4.0	1
290001 Secondary CTMT														
K/A Category Point Totals:	1	2	2	2	0	2	2	5	4	1	2	Group Point Total:		23

ES-401		BWR SRO Examination Outline Plant Systems - Tier 2/Group 2											Form ES-401-1	
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201001 CRD Hydraulic														
201002 RMCS														
201004 RSCS														
201006 RWM														
202001 Recirculation										9		Reactor Water Level	3.7	1
204000 RWCU														
205000 Shutdown Cooling											2.1.32	Ability To Explain And Apply System Limits And Precautions	3.8	1
214000 RPIS														
215002 RBM														
215003 IRM														
219000 RHR/LPCI: Torus/Pool Cooling Mode				8								Adequate Pump NPSH	3.0	1
230000 RHR/LPCI: Torus/Pool Spray Mode														
234000 Fuel Handling Equipment	6											RC&IS	3.2	1
239003 MSIV Leakage Control														
245000 Main Turbine Gen. and Auxiliaries										14		Generator Megavar Output	2.5	1
259001 Reactor Feedwater											2.4.48	Ability To Interpret Control Room Indications To Verify The Status And Operation Of System / And Understand How Operator Actions And Directives Affect Plant And System Conditions	3.8	1
262002 UPS (AC/DC)									1			Transfer From Preferred To Alternate Source	3.1	1
263000 DC Electrical Distribution			2									Components Using DC Control Power (i.e Breakers)	3.8	1
271000 Offgas	7											Plant Air Systems	2.7	1
272000 Radiation Monitoring														
286000 Fire Protection									1			Fire Water Pump Start	3.4	1
290003 Control Room HVAC					3							Temperature Control	2.7	1
300000 Instrument Air								1				Air Dryer And Filter Malfunctions	2.8	1
400000 Component Cooling Water				1								Automatic Start Of Standby Pump	3.9	1
K/A Category Point Totals:	2	0	1	2	1	0	0	1	2	2	2	Group Point Total:		13

ES-401		BWR SRO Examination Outline Plant Systems - Tier 2/Group 3										Form ES-401-1		
System # / Name	K1	K2	K3	K4	K5	K6	A1	A2	A3	A4	G	K/A Topic(s)	Imp.	Points
201003 Control Rod and Drive Mechanism			1									Reactor Power	3.4	1
215001 Traversing In-core Probe														
233000 Fuel Pool Cooling and Cleanup					1							Heat Removal Mechanisms	2.7	1
239001 Main and Reheat Steam														
256000 Reactor Condensate														
268000 Radwaste														
288000 Plant Ventilation									1			Isolation/Initiation Signals	3.8	1
290002 Reactor Vessel Internals								5				Exceeding Thermal Limits	4.2	1
K/A Category Point Totals:	0	0	1	0	1	0	0	1	1	0	0	Group Point Total:		4
Plant-Specific Priorities														
System / Topic	Recommended Replacement for...						Reason						Points	
295001 / G2.4.11	295001 AK2.02						ONI-C51 Revision (Post RF08)						1	
Plant-Specific Priority Total (limit 10):													1	

Facility: Perry		Date of Exam: July 2001		Exam Level: SRO	
Category	K/A #	Topic	Imp.	Points	
Conduct of Operations	2.1.6	Ability To Supervise And Assume A Management Role During Plant Transients And Upset Conditions	4.3	1	
	2.1.12	Ability To Apply Technical Specifications For A System	4.0	1	
	2.1.29	Knowledge Of How To Conduct And Verify Valve Lineups	3.3	1	
	2.1.3	Knowledge Of Shift Turnover Practices	3.4	1	
	2.1.20	Ability To Execute Procedure Steps	4.2	1	
	Total			5	
Equipment Control	2.2.27	Knowledge Of The Refueling Process	3.5	1	
	2.2.23	Ability To Track Limiting Conditions For Operations	3.8	1	
	2.2.18	Knowledge Of The Process For Managing Maintenance Activities During Shutdown Operations	3.6	1	
	2.2.6	Knowledge Of The Process For Making Changes In Procedures As Described In The Safety Analysis Report	3.3	1	
	Total			4	
Radiation Control	2.3.2	Knowledge Of Facility ALARA Program	2.9	1	
	2.3.4	Knowledge Of Radiation Exposure Limits / Including Permissible Levels In Excess Of Those Authorized	3.1	1	
	Total			2	
Emergency Procedures/ Plan	2.4.1	Knowledge Of EOP Entry Conditions And Immediate Action Steps	4.6	1	
	2.4.5	Knowledge Of The Organization Of The Operating Procedures Network For Normal / Abnormal / And Emergency Evolutions	3.6	1	
	2.4.15	Knowledge Of Communications Procedures Associated With EOP Implementation	3.5	1	
	2.4.12	Knowledge Of General Operating Crew Responsibilities During Emergency Operations	3.9	1	
	2.4.9	Knowledge Of Low Power / Shutdown Implications In Accident Mitigation Strategies	3.9	1	
	2.4.24	Knowledge Of Loss Of Cooling Water Procedures	3.7	1	
	Total			6	
Tier 3 Point Total (RO/SRO)				13/17	

**NRC WRITTEN EXAMINATION JUNE 2001**  
**PERRY NUCLEAR POWER PLANT**  
**Revision 2**

**PROCEDURES AND STANDARDS**

OPERATOR TRAINING GUIDELINE

OTG-18, NRC INITIAL EXAM DEVELOPMENT, VALIDATION AND  
ADMINISTRATION

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

NUREG 1021 REVISION 8, SUPPLEMENT 1, OPERATOR LICENSING  
EXAMINATION STANDARDS FOR POWER REACTORS

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

**NRC WRITTEN EXAMINATION JUNE 2001**  
**PERRY NUCLEAR POWER PLANT**  
**Revision 2**

**JUSTIFICATIONS FOR DELETIONS ON**  
**WRITTEN EXAMINATION OUTLINE**

**EMERGENCY PLANT EVOLUTIONS (EPEs) DELETED (BOTH SRO AND RO)**

295035      Secondary Containment High Differential Pressure - This is not an entry condition for PEI-N11, Secondary Containment Control. Perry does not have a true Secondary Containment (See System 290001 deletion justification). A high radiation release would be covered by 295038 (High Off-Site Release Rate), 295034 (Sec Cont Vent Rad High), or 295033 (Sec Cont Area Rad Levels).

**SYSTEMS DELETED (BOTH SRO AND RO)**

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

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

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

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

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

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

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

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

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

**NRC WRITTEN EXAMINATION JUNE 2001**  
**PERRY NUCLEAR POWER PLANT**  
**Revision 2**

- 239003 Main Steam Isolation Valve Leakage Control System – This system has been deleted as authorized by Tech Spec Amendment 103.
- 268000 Radwaste System – Radwaste Systems are operated by personnel in the Radwaste/Environmental/Chemistry Section (RECS); this is not a job function of the licensed Control Room operators at Perry.
- 215001 Traversing In-Core Probe System – This system is operated by Reactor Engineering personnel; this is not a job function of the licensed Control Room operators at Perry.
- 290001 Secondary Containment System – Contacted AM Stone (NRC) on a previous NRC Written Exam with information from our USAR (Section 6.2.3), PEI Bases Document, and Technical Specifications (LCO 3.6.4.1) with system design information in order to make a comparison with the K/As for this system in order to justify its deletion from the written sample plan. Based on this information provided, this system has been deleted from the random sample plan.

**GENERIC CATEGORIES DELETED (BOTH SRO AND RO)**

- GEN 2.2.3 Category is applicable to multi-unit facilities; Perry is a single unit facility.
- GEN 2.2.4 Category is applicable to multi-unit facilities; Perry is a single unit facility.

**WRITTEN (SRO ONLY)**

- Tier 1 Group 2 Evolutions 295028, and 600000 randomly deleted.
- Tier 2 Group 2 Systems 272000, 215003, 204000, and 201001 randomly deleted.
- Tier 2 Group 3 Systems 239001 and 256000 randomly deleted.
- 295024 EK2.19 / EK2.08 / EK2.01 not applicable to Perry.
- 295022 AK2.06 not applicable to Perry.
- 500000 EK3 has only one topic applicable to Perry (EK3.03).
- 209001 A1.06 Importance Value <2.5.
- 259002 K4.16 not applicable to Perry.
- 290002 A1 has no topics.

**NRC WRITTEN EXAMINATION JUNE 2001**  
**PERRY NUCLEAR POWER PLANT**  
**Revision 2**

215002 A4.02 not applicable to Perry.

261000 K5 has no topics applicable to Perry.

Tier 3 Gen 2.3.6 had to be deleted due to the limited scope of the topic which resulted in the inability to produce an SRO-only question with three viable distractors. Received permission from M. Bielby on 4/18/01 to randomly select another topic in Generic Category 2.3.

**WRITTEN (RO ONLY)**

Tier 1 Group 2 Evolutions 295019, 295020, 295028, and 600000 were randomly deleted.

500000 EK3 has only one topic applicable to Perry (EK3.03).

201001 A1.08 Importance Value <2.5.

215004 A2.06 Importance Value <2.5.

Tier 3 Gen 2.3.3 / 2.3.5 / 2.3.6 / 2.3.8 Importance Value <2.5

272000 K5.01 had to be deleted due to the delayed implementation of the Hydrogen Water Chemistry System (P73). The date for initial system operation is unknown, therefore, this topic cannot be included in the outline. Received permission from M. Bielby on 3/28/01 to randomly select another K/A for this system.

**WRITTEN EXAMINATION**

**Plant Generics Knowledge and Abilities**

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

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



# **Packing List**

## **Perry NRC Initial Written Examination**

### **2001**

#### **Blue Folder #1**

Tab I	Packing List
Tab B	ES-401-7
Tab C	ES 401-1 & ES 401-5 SRO Written Exam Outline
Tab D	ES 401-1 & ES 401-5 RO Written Exam Outline
Tab E	Justification Memo
Tab F	NRC Written Exam Question Tracking Sheet / Answer Key
Tab G	Master NRC RO Exam
Tab H	Master NRC RO Exam References
Tab IJ	Master NRC SRO Exam
Tab K	Master NRC SRO Exam References
Tabs L-Z	Empty

#### **Blue Folder #2**

Master Questions 1 – 35 (with references)

#### **Blue Folder #3**

Master Questions 36 – 70 (with references)

#### **Blue Folder #4**

Master Questions 71 – 105 (with references)

#### **Blue Folder #5**

Master Questions 106 – 131 (with references)

**\*Refer to the NRC Written Exam Question Tracking Sheet to cross-reference Master Questions 1 – 127 with their respective RO and SRO Master Exams.**

**U.S. Nuclear Regulatory Commission  
Site-Specific  
Written Examination**

**Applicant Information**

Name:	Region: I / II / III / IV
Date:	Facility/Unit: <b>Perry</b>
License Level: <b>RO</b> / SRO	Reactor Type: W / CE / BW / <b>GE</b>
Start Time:	Finish Time:

**Instructions**

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

**Applicant Certification**

All work done on this examination is my own. I have neither given nor received aid.

\_\_\_\_\_  
Applicant's Signature

**Results**

Examination Value	100.00	Points
Applicant's Score	_____	Points
Applicant's Grade	_____	Percent

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 1

The plant is operating at 100% power. The following equipment is in operation:

- CRDH Pump 'A'
- NCC Pump 'A' and 'B'

A loss of power to Bus EH11 occurs when the Preferred Source Breaker trips open.

Which one of the following describes the expected response of circuit breakers associated with Bus EH11?

- A. Bus EH11 Stub Bus Breaker opens and NCC Pump 'A' breaker opens.
- B. Bus EH11 Stub Bus Breaker opens and CRDH Pump 'A' breaker remains closed.
- C. Bus EH11 Stub Bus Breaker remains closed and CRDH Pump 'A' breaker opens.
- D. Bus EH11 Stub Bus Breaker remains closed and NCC Pump 'A' breaker remains closed.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 2

Prior to a manual reactor scram per IOI-8, Shutdown By Manual Reactor Scram, the plant is operating at 100% power with the Feedwater Master Level Controller tape set at 200 inches.

Which one of the following describes the response of the Feedwater Level Control System following the manual reactor scram signal?

Assume no operator actions are taken.

- A.           Upon receipt of the manual reactor scram signal, the level demand signal will be 196 inches for 10 seconds and then decrease to 178 inches.
- B.           Upon receipt of the manual reactor scram signal, the level demand signal will be 200 inches for 10 seconds and then decrease to 178 inches.
- C.           When RPV water level decreases to 178 inches, the level demand signal will be 196 inches for 10 seconds and then decrease to 178 inches.
- D.           When RPV water level decreases to 178 inches, the level demand signal will be 200 inches for 10 seconds and then decrease to 178 inches.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 3

The plant is operating at 95% power.

- SB&PR Channel 'A' is in service
- SB&PR Channel 'B' is in TEST

The main steam pressure transmitter output signal to SB&PR Channel 'A' fails upscale.

Which one of the following describes the valve response associated with the SB&PR System?

Assume no operator actions are taken.

- A. Turbine Control Valves and Bypass Valves open.
- B. Turbine Control Valves and Bypass Valves remain 'as-is'.
- C. Turbine Control Valves remain 'as-is' and Bypass Valves open.
- D. Turbine Control Valves open and Bypass Valves remain closed.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 4

Which one of the following correctly lists the signal(s) which will automatically close the Containment isolation valves for the Drywell Equipment and Floor Drain Sump Systems?

- A. High Drywell pressure (1.68 psig) or low Reactor water level (L1).
- B. High Drywell pressure (1.68 psig) or low Reactor water level (L2).
- C. Drain sump high discharge temperature.
- D. Drain sump pump high discharge pressure.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 5

An ATWS is in progress.

PEI-B13, RPV Control (ATWS) directs the Control Room Operators to “Stabilize RPV pressure to less than 1000 psig”.

Which one of the following conditions meets the intent of this PEI action step?

- A. Automatic operation of the Bypass Valves.
- B. Automatic operation of the SRVs on Low-Low Setpoint.
- C. Uncontrolled depressurization due to a stuck-open Bypass Valve.
- D. Manual operation of SRVs when sufficient Bypass Valve capacity exists.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 6

During an incomplete scram, which one of the following Redundant Reactivity Control System (RRCS) control signals requires an 'APRM Not Downscale' permissive?

- A. SLC Pump trip
- B. Feedwater Runback (FWRB)
- C. Alternate Rod Insertion (ARI)
- D. Recirc Pump Transfer to LFMG



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 7

ONI-C61, Evacuation of the Control Room, has been entered.

The reactor was not scrammed prior to leaving the Control Room.

Which one of the following is the preferred method to fully insert all control rods from outside the Control Room?

- A. Open the specified scram air header drain valves.
- B. Open the specified RPS MG set output breakers.
- C. Cycle the specified RPS power distribution panel breakers.
- D. Cycle the specified ATWS UPS distribution panel breakers.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 8

The FHB Ventilation System is in operation in accordance with SOI-M40. During movement of irradiated fuel in the FHB, an irradiated fuel bundle is dropped.

Shortly thereafter, a HIGH radiation alarm is received on the FHB Ventilation Exhaust GAS module.

Which one of the following describes the current status of the FHB Ventilation System?

- A.            No Exhaust Fan running; no Supply Fan running.
- B.            No Exhaust Fan running; one Supply Fan running.
- C.            Two Exhaust Fans running; no Supply Fan running.
- D.            Two Exhaust Fans running; one Supply Fan running.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 9

The plant has experienced a Loss of Coolant Accident due to a complete break of the Recirculation System piping.

Which one of the following describes the effects of Drywell pressure?

Drywell pressure will rise to a maximum value, thereby \_\_\_\_\_.

- A. clearing the Drywell to Containment Horizontal Vents, releasing steam directly into the Containment and pressurizing Containment to a maximum value.
- B. clearing the Drywell to Containment Horizontal Vents and causing a rise in Containment pressure followed by a lowering of Drywell pressure and recovering of the vents.
- C. covering the Drywell to Containment Horizontal Vents and preventing a rise in Containment pressure.
- D. preventing the uncovering of the Drywell to Containment Horizontal Vents and preventing a rise in Containment pressure.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 10

Which one of the following parameters will require the initiation of a reactor scram in accordance with the Plant Emergency Instructions (PEIs)?

Assume that the reactor is operating and no ATWS condition exists.

- A. Drywell pressure is 1.2 psig.
- B. Suppression Pool level is 14.50 ft.
- C. Containment temperature is 185°F.
- D. Suppression Pool temperature is 97°F.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 11**

Why is a level of 7.25 feet in the Suppression Pool a concern to the Control Room Operators when operating in the PEIs?

- A.            Operation at this level could cause air entrapment at the RCIC suction strainer.
- B.            Operation at this level will uncover the Suppression Pool suction strainer.
- C.            Operation at this level could result in exceeding the stress limits of the SRV tail pipe.
- D.            Operation at this level will cause rapid pressurization of Containment during an SRV lift.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 12

The plant is operating at 50% power with both Reactor Feed Pump Turbines (RFPTs) on the Master Level Controller (MLC).

Which one of the following describes the expected response of the Feedwater System if the Narrow Range level instrument input to the MLC fails upscale?

Assume no operator actions are taken?

- A. Feedwater flow increases; both RFPTs trip on mechanical overspeed.
- B. Feedwater flow increases; both RFPTs trip on high RPV level (L8).
- C. Feedwater flow decreases; both RFPTs trip on low RPV level (L2).
- D. Feedwater flow decreases; both RFPTs trip on RCIC initiation.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 13

Which one of the following describes the reactor core conditions that Cold Shutdown Boron Weight is based upon in order to provide an adequate shutdown margin?

- A. 70°F fuel/moderator temperature, xenon free, and all control rods fully withdrawn.
- B. 70°F fuel/moderator temperature, xenon free, and 50% rod density.
- C. 68°F fuel/moderator temperature, xenon free, and all control rods fully withdrawn.
- D. 68°F fuel/moderator temperature, xenon free, and 50% rod density.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 14

PEI-D17, Radioactivity Release Control contains the following action step:

“Except for systems required to assure adequate core cooling or shutdown the reactor, isolate all primary systems that are discharging into areas outside one or more of the following: Annulus, Auxiliary Building, Intermediate Building, or Steam Tunnel.”

Which one of the following defines the term “primary system” as used in PEI-D17?

A “primary system” refers to any system \_\_\_\_\_.

- A. that can be used to reduce RPV pressure.
- B. that can be used to maintain RPV water level.
- C. connected to the RPV and contains radioactive coolant.
- D. connected to the RPV and will have a reduced leak rate if RPV pressure is lowered.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 15

The following plant conditions exist:

- A Loss of Coolant Accident has occurred
- Hydrogen is present in the Primary Containment
- PEI-M51/56, Hydrogen Control, has been entered
- Hydrogen Recombiners have been started

Which one of the following hydrogen concentrations will require the Hydrogen Recombiners to be secured, including the bases for this action?

The Hydrogen Recombiners are secured at \_\_\_\_\_.

- A. 4% hydrogen concentration in order to prevent their becoming an ignition source.
- B. 4% hydrogen concentration because there is insufficient oxygen available to support the recombination reaction.
- C. 6% hydrogen concentration because there is insufficient oxygen available to support the recombination reaction.
- D. 6% hydrogen concentration in order to prevent their becoming an ignition source.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 16

The plant is operating at 100% power. Reactor Narrow Range Level Channel 'B' is selected for input to the Master Level Controller. DC Bus D-1-A de-energizes due to a fault condition.

Which one of the following describes the consequences of the failure to perform the immediate actions of ONI-R42-4, Loss of DC Bus D-1-A?

- A.           RPV water level will increase causing a reactor scram but RFPT 'A' and the Main Turbine will not automatically trip on high water level.
- B.           RPV water level will increase until the RFPTs trip on high water level and the reactor will scram when water level decreases to RPV Level 3.
- C.           RPV water level will increase until the RFPTs trip on high water level but a reactor scram will not occur due to the loss of DC Bus D-1-A.
- D.           RPV water level will increase and then stabilize at a higher level when the level error signal overcomes the flow error signal to RFPT 'A'.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 17**

Which one of the following describes the effect on the Reactor Recirculation System for an actuation of the End of Cycle Recirculation Pump Trip (EOC-RPT) logic following a Main Turbine trip?

- A. At greater than 38% rated thermal power, only the CB5 breaker will trip, the Low Frequency Motor Generator (LFMG) will start, and the CB2 breaker will close for each Recirculation Pump.
- B. At greater than 38% rated thermal power, the CB3, CB4, and CB5 breakers will trip, the Low Frequency Motor Generator (LFMG) will start, and the CB2 breaker will close for each Recirculation Pump.
- C. At less than 38% rated thermal power, only the CB5 breaker will trip, the Low Frequency Motor Generator (LFMG) will start, and the CB2 breaker will close for each Recirculation Pump.
- D. At less than 38% rated thermal power, the CB2, CB3, and CB4 breakers will trip for each Recirculation Pump.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 18

The High RPV Water Level Trip (L8) is designed to prevent \_\_\_\_\_.

- A. RCIC Turbine damage due to water slugging.
- B. MSL hanger damage due to excessive weight.
- C. Main Turbine damage due to moisture carryover.
- D. MSIV damage due to excessive hydraulic loading.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 19

Which one of the following describes the bases for maximizing Containment cooling during the execution of PEI-T23, Containment Control?

- A. To prevent exceeding the Containment design temperature limit of 330°F.
- B. To prevent exceeding the Containment average air temperature LCO limit of 145°F.
- C. To prevent exceeding the environmental qualification temperature of 330°F for safety-related electrical equipment in the Containment.
- D. To prevent exceeding the environmental qualification temperature of 185°F for safety-related electrical equipment in the Containment.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 20

The plant is operating at 100% power.

- TBCC HX OUTLET TEMP HIGH alarm is received on panel H13-P870
- ONI-P44, Loss of Turbine Building Closed Cooling, has been entered
- TBCC Heat Exchanger Outlet Temperature Control Valve, 1P41-F003, was confirmed to have failed in the 'close' position

Which one of the following describes the correct action to perform in accordance with ONI-P44?

- A. Contact I&C to troubleshoot the temperature controller for 1P41-F003.
- B. Direct a Non-Licensed Operator to manually throttle open 1P41-F003.
- C. Open TBCC Heat Exchanger Service Water Temperature Control Valve Bypass Valve, 1P41-F390.
- D. Shutdown a Service Water Pump per SOI-P40/41 until only one Service Water Pump is operating.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 21

The plant is in MODE 2 when the running CRDH Pump trips.

CRD charging water header pressure decreases to reactor pressure.

Which one of the following describes the plant conditions that would require immediately placing the Reactor Mode Switch in the SHUTDOWN position in accordance with ONI-C11-1, Inability To Move Control Rods?

- A.           Reactor pressure is 500 psig.  
              Accumulator fault occurs on control rod 20-27 at position 00.
- B.           Reactor pressure is 500 psig.  
              Accumulator fault occurs on control rod 20-27 at position 24.
- C.           Reactor pressure is 700 psig.  
              Accumulator fault occurs on control rod 20-27 at position 00.
- D.           Reactor pressure is 700 psig.  
              Accumulator fault occurs on control rod 20-27 at position 24.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 22

Which one of the following Plant Emergency Instruction (PEI) curves or limits, if exceeded, could directly result in a loss of Primary Containment integrity?

- A. Primary Containment Limit (PCL).
- B. Pressure Suppression Pressure (PSP).
- C. SRV Tail Pipe Level Limit (SRVTPLL).
- D. Maximum Core Uncovery Time Limit (MCUTL).



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 23

Which one of the following describes the bases for isolating any system discharging into the annulus and surrounding containment in accordance with PEI-N11, Containment Leakage Control?

- A. To terminate rising temperatures, radiation levels, and water levels in the Secondary Containment.
- B. To protect equipment in the Annulus and Primary Containment.
- C. To minimize reactor coolant inventory loss.
- D. To preserve Turbine Building accessibility.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 24

Which one of the following alarms, if received on AB EL 574' EAST area radiation monitor, would require entry into PEI-N11, Containment Leakage Control?

- A. Fail
- B. Alert
- C. High
- D. High-High

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 25

The following plant conditions exist:

- A steam leak has occurred in the RCIC Pump Room
- HIGH radiation alarm is received on the Auxiliary Building Ventilation Exhaust GAS module
- Auxiliary Building Ventilation Exhaust GAS module indicates 29,000 cpm
- PEI-N11, Containment Leakage Control, has been entered

Which one of the following describes the correct action to perform in accordance with PEI-N11?

**PEI-N11 flowchart is provided for reference.**

Direct a Non-Licensed Operator to \_\_\_\_\_.

- A. start a second Auxiliary Building Ventilation Supply Fan.
- B. verify the Auxiliary Building Ventilation Supply Fan is tripped.
- C. secure the Auxiliary Building Ventilation Supply and Exhaust Fans.
- D. verify the Auxiliary Building Ventilation Supply and Exhaust Fans are running.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 26

The following plant conditions exist:

- A 25 gpm leak has occurred on the ESW-side of the RHR 'A' Heat Exchanger
- RHR A PUMP ROOM SUMP LEVEL HIGH alarm is received on panel H13-P601

Which one of the following describes how this leak is captured by the Liquid Radwaste Sumps System (G61)?

Assume no operator actions have been taken.

- A. The RHR 'A' Pump Room sump drain valve will automatically open to route the leakage to the Auxiliary Building Floor Drain Sump.
- B. The RHR 'A' Pump Room sump drain valve will automatically close to contain the leakage to the RHR 'A' Pump Room.
- C. The RHR 'A' Pump Room sump will gravity drain to the Auxiliary Building Floor Drain Sump.
- D. The RHR 'A' Pump Room sump will fill and overflow into the RHR 'A' Pump Room.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 27

During a reactor startup, control rods are being withdrawn to achieve 50% rod density.

- Rod Sequence 'A' is selected
- All Group 5 through 10 control rods are fully inserted
- All Group 1 and 2 control rods have been fully withdrawn
- All Group 3 control rods have been withdrawn to notch position '12'

The Control Room Operator selects a Group 4 control rod for withdrawal.

Which one of the following describes the response of the Rod Control and Information System (RC&IS)?

- A. Withdrawal of the control rod is immediately blocked.
- B. Withdrawal of the control rod will occur with no restrictions.
- C. Withdrawal of the control rod will be blocked when the control rod reaches notch position '04'.
- D. Withdrawal of the control rod will be blocked when the control rod reaches notch position '12'.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 28

The plant is operating at 100% power.

- AFDL IN CONTROL alarm is received on panel H13-P680
- APRM Channel 'A' failed upscale
- Fifteen (15) seconds later, APRM Channel 'A' is inadvertently bypassed

Assume no additional operator actions are taken.

Which one of the following describes the response of the Reactor Recirculation System Flow Control Valves (FCVs)?

- A. The Reactor Recirculation Loop Flow Controllers will return the FCVs to their pre-transient valve positions.
- B. The Reactor Recirculation Loop Flow Controllers will maintain the FCVs at their current valve positions.
- C. The Automatic Flow Demand Limiter will return the FCVs to their pre-transient valve positions.
- D. The Automatic Flow Demand Limiter will maintain the FCVs at their current valve positions.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 29

Which one of the following describes the operation of the RHR PUMP 'A' & HX COOLER, M39-B001A?

The RHR PUMP 'A' & HX COOLER will automatically start \_\_\_\_\_.

- A.           when an RHR LOCA initiation signal seals-in (K110A relay contact); room heat is dissipated by recirculating room air through a heat exchanger cooled directly by ECC.
- B.           when an RHR LOCA initiation signal seals-in (K110A relay contact); room heat is dissipated by recirculating room air through a heat exchanger cooled directly by ESW.
- C.           when RHR Pump 'A' breaker closes (52a contact); room heat is dissipated by recirculating room air through a heat exchanger cooled directly by ECC.
- D.           when RHR Pump 'A' breaker closes (52a contact); room heat is dissipated by recirculating room air through a heat exchanger cooled directly by ESW.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 30

One hour ago the Control Room Operators discovered that the blue pressure permissive light for the LPCS Injection Valve, 1E21-F005, was not lit.

Control Room Operators confirmed the blue light bulb was good.

Which one of the following describes the operation of the LPCS Injection Valve control logic if a Loss-of Coolant Accident occurs?

- A. The LPCS Injection Valve automatically opens, irrespective of RPV pressure, due to the LPCS LOCA initiation signal.
- B. The LPCS Injection Valve remains closed and cannot be opened with its control switch until RPV pressure decreases to 600 psig.
- C. The LPCS Injection Valve remains closed and cannot automatically open until RPV pressure decreases to 530 psig.
- D. The LPCS Injection Valve remains closed and cannot automatically open until RPV pressure decreases to 600 psig.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 31

The following plant conditions exist:

- A Loss of Coolant Accident has occurred
- Drywell pressure is 1.8 psig
- RPV water level is +195 inches and stable
- The High Pressure Core Spray (HPCS) Pump has been overridden to STOP

Subsequently, Bus EH13 loses power and is re-energized by the HPCS Diesel Generator.

Assume no additional operator actions were taken.

Which one of the following describes the current condition of the HPCS Pump?

The HPCS Pump is \_\_\_\_\_.

- A. not running because the initiation logic was reset.
- B. not running because the override logic was not affected.
- C. running because the override logic was reset.
- D. running because the initiation logic was not affected.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 32

The following plant conditions exist:

- MODE 5
- RPS INST VOL HI annunciator on panel H13-P680 is in alarm
- All INST VOL LEVEL HI SCRAM BYPASS switches are in BYPASS
- RPS logic is reset

Which one of the following describes the effect on the Reactor Protection System (RPS) when the Reactor Mode Switch is taken from SHUTDOWN, through REFUEL, to the STARTUP/STANDBY position?

- A. RPS actuation occurs because the Scram Discharge Volume High trip is enabled when the Reactor Mode Switch is in the STARTUP/STANDBY position.
- B. RPS actuation occurs because the Scram Discharge Volume High trip is enabled when the Reactor Mode Switch is in the REFUEL position.
- C. RPS actuation does not occur because the Scram Discharge Volume High trip is bypassed when the Reactor Mode Switch is in the STARTUP/STANDBY position.
- D. RPS actuation does not occur because the Scram Discharge Volume High trip is bypassed when the INST VOL LEVEL HI SCRAM BYPASS switches are in BYPASS.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 33

The plant is operating at 75% power.

Which one of the following setpoints is displayed when the Operator at the Controls depresses the APRM ALARM LEVEL RECORD pushbutton for an IRM/APRM Recorder?

- A. Fixed neutron flux upscale rod block
- B. Fixed neutron flux upscale scram
- C. Flow biased rod block
- D. Flow biased scram

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 34

RHR Loop 'B' is operating in the shutdown cooling mode when a loss of RPS Bus 'A' occurs.

Which one of the following describes the response of RHR Loop 'B'?

- A. RHR Loop 'B' remains in operation.
- B. Only an RHR SDC Inboard isolation occurs.
- C. Only an RHR SDC Outboard isolation occurs.
- D. An RHR SDC Inboard and Outboard isolation occurs.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 35

The following plant conditions exist:

- A Station Blackout (SBO) is in progress
- RCIC System is maintaining RPV water level
- RCIC suction is from the CST

Which one of the following describes the response of the RCIC suction valve(s) if a high Suppression Pool (SP) level occurs?

- A. The RCIC suction valves fail 'as-is'.
- B. The SP suction valve remains closed.
- C. The CST suction valve remains open.
- D. The RCIC suction valve transfer occurs.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 36**

Which one of the following is the power source to the ADS 'B' solenoid valves?

- A. ED-1-A
- B. ED-1-B
- C. D-1-A
- D. D-1-B

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 37**

The following plant conditions exist:

- A Loss of Coolant Accident has occurred
- Hydrogen is present in the Drywell
- PEI-M51/56, Hydrogen Control, has been entered
- Combustible Gas Mixing Compressors have been started

Which one of the following conditions would cause Combustible Gas Mixing Compressor 'A' to trip?

- A. RHR Pump 'A' control switch taken to STOP.
- B. Less than 17 gpm cooling water flow to the oil cooler.
- C. Less than 80 gpm cooling water flow to the air aftercooler.
- D. LPCI A Injection Valve, 1E12-F042A, less than 90% open.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 38**

The following plant conditions exist:

- A Loss of Coolant Accident has occurred
- PEI-M51/56, Hydrogen Control, has been entered
- Hydrogen Igniters were started
- Containment hydrogen concentration has exceeded HDOL

Which one of the following conditions would require the Hydrogen Igniter control switches to be placed in the OFF position?

- A. A loss of power occurs to the Hydrogen Analyzers.
- B. A loss of power occurs to the Hydrogen Igniters.
- C. An oxygen-inert environment exists in Containment.
- D. A steam-inert environment exists in Containment.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 39

Following a reactor scram, RPV water level decreased to +100 inches.

Which one of the following systems will have automatically isolated?

- A. Nuclear Closed Cooling Water System (P43)
- B. Safety-Related Instrument Air System (P57)
- C. Reactor Water Cleanup System (G33)
- D. Fire Service Water System (P54WTR)

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 40

The following plant conditions exist:

- RHR loop 'A' started in the LPCI mode on a high Drywell pressure signal
- RHR Pump 'A' was shutdown by taking its control switch to the STOP position

Which one of the following conditions will automatically re-start RHR Pump 'A'?

Assume no further operator actions are taken.

- A. Automatic Depressurization System (ADS) initiation signal occurs.
- B. Drywell pressure signal clears and re-occurs.
- C. Containment Spray initiation signal occurs.
- D. RPV low water level (L1) signal occurs.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 41

An SRV tailpipe vacuum breaker has failed in the 'open' position during SRV operations.

Which one of the following conditions will result, potentially requiring entry into PEI-T23, Containment Control?

- A. Water will be drawn up into the tailpipe from the Suppression Pool each time an SRV is closed.
- B. Steam will bypass the quencher and discharge directly into the Suppression Pool each time an SRV is opened.
- C. Steam will discharge from the tailpipe directly into the Containment airspace each time an SRV is opened.
- D. Steam will discharge from the tailpipe directly into the Drywell airspace each time an SRV is opened.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 42**

The following plant conditions exist:

- Main Turbine roll is in progress
- STARTING RATE – MEDIUM has been selected
- SPEED SET RPM – 1800 has been selected
- Main Turbine speed is 500 rpm and increasing

Which one of the following is the effect on the acceleration rate of the Main Turbine if one of the acceleration input signals in the Speed Control Unit is lost?

The acceleration rate will \_\_\_\_\_.

- A. decrease by  $1/3$  (one-third).
- B. decrease by  $1/2$  (one-half).
- C. remain the same.
- D. double.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 43**

Which one of the following sets of Feedwater controls provides a speed demand signal to the Reactor Feed Pump Turbines?

**Note: A partial list will be incorrect.**

- A. Manual Speed Control Dial, Startup Level Controller, and Master Level Controller.
- B. Low Flow Controller, Manual Speed Control Dial, and Startup Level Controller.
- C. Startup Level Controller, Low Flow Controller, and Master Level Controller.
- D. Master Level Controller, Manual Speed Control Dial, and Low Flow Controller.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 44

The plant is operating at 40% power.

- REACTOR NARROW RANGE LEVEL Switch, 1C34-S1, is selected to Channel 'A'
- Reactor Narrow Range Channel 'A' indicates +196 inches on 1C34-R606A
- Reactor Narrow Range Channel 'B' indicates +193 inches on 1C34-R606B
- The Master Level Controller is in operation with its tapeset at +196 inches

Which one of the following describes the RPV water level response, as indicated on Reactor Narrow Range Channel 'B', when the Control Room Operator switches 1C34-S1 to Channel 'B'?

RPV water level initially \_\_\_\_\_.

- A. decreases and then stabilizes at +193 inches.
- B. decreases and then stabilizes at +196 inches.
- C. increases and then stabilizes at +196 inches.
- D. increases and then stabilizes at +199 inches.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 45

A Main Turbine/Generator trip occurred and station loads failed to transfer to the Unit 1 Startup Transformer.

Which one of the following Immediate Actions is required by ONI-N32, Turbine And/Or Generator Trip?

- A. Close START-UP SUPPLY BRKRS L1006 and L1009.
- B. Open NORMAL SUPPLY BRKRS, L1102 and L1202.
- C. Enter ONI-R10, Loss Of AC Power.
- D. Enter ONI-R22-1, Loss Of An Essential And / Or A Stub 4.16 KV Bus.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 46

A Loss of Off-Site Power (LOOP) occurs.

RPV water level decreases to +10 inches.

Ten minutes later, the Division 1 DG jacket water pump fails.

Which one of the following describes the effect on Bus EH11 due to the Division 1 DG jacket water pump failure?

- A. Bus EH11 de-energizes because the Division 1 DG trips on high jacket water temperature.
- B. Bus EH11 de-energizes because the Division 1 DG trips on high lube oil temperature.
- C. Bus EH11 remains energized because the Division 1 DG high jacket water temperature trip is bypassed on a LOOP signal.
- D. Bus EH11 remains energized because the Division 1 DG high jacket water temperature trip is bypassed on a LOCA signal.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 47

If a Reactor Recirculation Pump is started at a reduced RPV water level, then RPV water level in the downcomer annulus will initially \_\_\_\_\_.

- A. decrease which may cause an RPS actuation due to RPV Level 3.
- B. decrease which may cause cavitation of the Flow Control Valves.
- C. increase which may cause an RPS actuation due to high neutron flux.
- D. increase which may cause an excessive cooldown of the RPV.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 48

Which one of the following requirements provides for adequate NPSH for RHR Pump 'A' during startup or operation in the Suppression Pool Cooling mode?

- A. The LPCS & RHR A waterleg pump must be running before RHR Pump 'A' will start.
- B. The Suppression Pool suction valve must be open before RHR Pump 'A' will start.
- C. The LPCS System must not be operated in the LPCS Test mode during RHR 'A' Suppression Pool Cooling mode operation.
- D. The RHR System flow must be maintained greater than 6000 gpm during RHR 'A' Suppression Pool Cooling mode operation.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 49

A refueling outage is in progress. One control rod is selected and withdrawn with the Reactor Mode Switch in the REFUEL position.

Which one of the following Refueling Platform (F15) operations would be prevented?

- A. Removing a fuel assembly from the RPV with the Main Fuel Hoist.
- B. Removing a control rod blade from the RPV with the Auxiliary Hoist.
- C. Moving the Refueling Platform inside the RPV with the Main Fuel Hoist unloaded.
- D. Moving the Refueling Platform from IFTS to the RPV with the Main Fuel Hoist unloaded.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 50

The plant is operating at 100% power with the following Main Generator conditions:

- Electrical output            1000 MWe
- Hydrogen pressure        60 psig
- Power factor                1

Select the maximum allowable reactive load for the Main Generator under these conditions.

**PDB-C0002, Generator Capability Curve is provided for reference.**

- A.            575 MVars
- B.            675 MVars
- C.            700 MVars
- D.            775 MVars

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 51

Which one of the following conditions will cause the static transfer switch in the Plant Vital Balance of Plant uninterruptible power supply (BOP-UPS) system to automatically shift?

- A. Low cooling air flow to the inverter.
- B. High voltage sensed at the output of the inverter.
- C. Ground fault sensed on Vital Distribution Bus V-1-A.
- D. Loss of battery chargers to Battery D-1-A for more than 15 minutes.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 52

DC control power is lost to Bus H11.

Which one of the following describes the operation of circuit breakers supplying loads from Bus H11?

The circuit breakers can \_\_\_\_\_.

- A. be opened and closed locally at the breaker cubicle.
- B. be closed from the Control Room and opened locally at the breaker cubicle.
- C. be closed locally at the breaker cubicle, however, all circuit breaker automatic trip functions are still available.
- D. be opened and closed from the Control Room, however, all circuit breaker automatic trip functions are disabled.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 53

An air purge of the Off-Gas System has been established in accordance with SOI-N64/62, Off-Gas / Condenser Air Removal System, during a plant startup.

Which one of the following describes the source of purge air and the point of injection into the Off-Gas System?

- A. Instrument Air enters at the inlet to the recombiners.
- B. Instrument Air enters at the inlet to the adsorbers.
- C. Service Air enters at the inlet to the preheaters.
- D. Service Air enters at the inlet to the gas dryers.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 54

The Fire Service Water System is in Standby Readiness when a fire system initiation occurs. Fire main pressure decreases to 110 psig and then increases to 150 psig.

Which one of the following describes the current configuration of the Fire Service Water System?

Assume no operator actions were performed.

Fire Service Jockey Pump	Motor Fire Service Pump	Diesel Fire Service Pump
-----------------------------	----------------------------	-----------------------------

- |    |         |         |         |
|----|---------|---------|---------|
| A. | Running | Off     | Off     |
| B. | Off     | Running | Off     |
| C. | Running | Running | Running |
| D. | Off     | Off     | Running |



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 55

The Non-Licensed Operator reports that the guide vanes on the operating Control Complex Chiller (P47) have failed closed.

Which one of the following describes the potential impact of this condition?

Control Room air temperature will \_\_\_\_\_.

- A. decrease; personnel habitability limits may be exceeded.
- B. decrease; equipment temperature limits may be exceeded.
- C. increase; equipment humidity limits may be exceeded.
- D. increase; equipment temperature limits may be exceeded.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 56

A Non-Licensed Operator reports that the refrigeration unit for an in-service Instrument Air Dryer is not operating.

This condition, if left uncorrected, may result in the introduction of \_\_\_\_\_ into the Instrument Air System.

- A. foreign particles
- B. wear products
- C. hydrocarbons
- D. water

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 57

The following equipment conditions exist:

- All ECCS and RCIC are in Standby Readiness
- Division 1 and 2 Diesel Generators are in Standby Readiness
- HPCS Diesel Generator is in Secured Status

A reactor scram occurs and RPV water level decreases to +120 inches.

Two minutes later, which one of the following describes the configuration of the Emergency Service Water (ESW) System?

Assume no operator actions are performed.

	<u>ESW Pump A</u>	<u>ESW Pump C</u>
A.	Off	Off
B.	Running	Off
C.	Running	Running
D.	Off	Running

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 58

A reactor startup/heatup is in progress with IRMs on Range 7.

During a single notch withdrawal of control rod 30-31 from position 08 to 10, the collet fingers stick in the 'unlocked' position.

Which one of the following describes the effect of this control rod withdrawal event?

Control rod 30-31 will \_\_\_\_\_.

- A. settle at position 10; reactor power and heatup rate will stabilize.
- B. drift into the core; reactor power and heatup rate will decrease.
- C. drift out of the core; reactor power and heatup rate will increase.
- D. remain at position 08 until drive water header pressure has been increased sufficiently to free the collet fingers.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 59

The Fuel Pool Cooling and Cleanup (FPCC) System is in operation when a RHR LOCA initiation occurs.

Which one of the following describes the cooling water flowpath configuration for the FPCC heat exchangers?

- A. Nuclear Closed Cooling water flowpath automatically isolates; Emergency Service Water flowpath automatically lines up to provide cooling water flow.
- B. Nuclear Closed Cooling water flowpath automatically isolates; Emergency Service Water flowpath must be manually lined up to provide cooling water flow.
- C. Nuclear Closed Cooling water flowpath must be manually isolated; Emergency Service Water flowpath must be manually lined up to provide cooling water flow.
- D. Nuclear Closed Cooling water flowpath must be manually isolated; Emergency Service Water flowpath automatically lines up to provide cooling water flow.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 60**

Which one of the following Radiation Monitors will isolate the Containment Vessel and Drywell Purge System (M14) if a high radiation condition occurs?

- A. Drywell Atmosphere Radiation Monitor, 1D17-K670.
- B. Containment Atmosphere Radiation Monitor, 1D17-K680.
- C. Containment Ventilation Exhaust Radiation Monitor, 1D17-K609A-D.
- D. Containment & Drywell Purge Exhaust Radiation Monitor, 1D17-K660.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 61

Which one of the following describes the procedural requirements for performance of a Valve Lineup Instruction (VLI)?

- A.       The valves can be positioned in any order unless a specific order is specified by the Unit Supervisor.  
          The Independent Verifier can verify the valves in any order.
- B.       The valves can be positioned in any order unless a specific order is specified by the Unit Supervisor.  
          The Independent Verifier must verify the valves in the same order.
- C.       The valves shall be positioned in the order in which they appear unless specified otherwise by the Unit Supervisor.  
          The Independent Verifier must verify the valves in the same order.
- D.       The valves shall be positioned in the order in which they appear unless specified otherwise by the Unit Supervisor.  
          The Independent Verifier can verify the valves in any order.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 62

The following conditions exist:

- You are an on-coming Control Room Operator (SO/US/SS)
- This is your first day back on-shift after 10 days of time off

Which one of the following describes the requirement for reviewing the Daily and Standing Instructions?

The Daily and Standing Instructions should be reviewed for the previous \_\_\_\_\_.

- A. one day.
- B. two days.
- C. seven days.
- D. ten days.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 63

An Infrequently Performed Test or Evolution (IPTE) is in progress which will demonstrate the heat removal capability of the RHR heat exchangers.

The following plant conditions exist:

- Reactor power is being maintained at 5%
- RCIC is operating in the CST-to-CST mode in order to raise Suppression Pool temperature to  $100 \pm 2$  °F
- Suppression Pool temperature is 97 °F and slowly increasing

Which one of the following describes the entry requirements for PEI-T23, Containment Control, and the Required Action requirements for LCO 3.6.2.1, Suppression Pool Average Temperature?

- A. PEI-T23 is not required to be entered; the Required Actions for LCO 3.6.2.1 are not required to be performed.
- B. PEI-T23 is not required to be entered; the Required Actions for LCO 3.6.2.1 are required to be performed.
- C. PEI-T23 is required to be entered; the Required Actions for LCO 3.6.2.1 are not required to be performed.
- D. PEI-T23 is required to be entered; the Required Actions for LCO 3.6.2.1 are required to be performed.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 64**

A General Emergency is in progress.

Which one of the following individuals is directly charged with the command authority over all activities involving plant operations.

- A. EOF Emergency Coordinator
- B. TSC Operations Manager
- C. Shift Supervisor
- D. Unit Supervisor

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 65

I&C has been authorized to start SVI-C71-T0042B, Drywell High Pressure Channel B Functional for 1C71-N650B.

The Operator at the Controls has been notified that the following alarms will occur during performance of this SVI:

- |                                     |             |
|-------------------------------------|-------------|
| • RPS DW PRESS HI                   | P680-05A-B8 |
| • 1/2 SCRAM B/D                     | P680-05A-B9 |
| • RPS B & D OUT OF SERVICE          | P680-05A-C5 |
| • RPS B TRIP UNIT IN CAL/FAIL       | P680-06A-D7 |
| • NS4 INBD ISOLATION OUT OF SERVICE | P601-18A-A3 |
| • BOP ISOL DW PRESS HIGH            | P601-19A-A6 |

There are no other surveillances in progress.

Which one of the following describes the required action to be performed if annunciator 1/2 SCRAM A/C occurs during performance of this surveillance?

- A. The alarm is 'unexpected'; notification of the Unit Supervisor is not required.
- B. The alarm is 'unexpected'; notification of the Unit Supervisor is required.
- C. The alarm is 'expected'; alarm confirmation is not required.
- D. The alarm is 'expected'; alarm confirmation is required.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 66**

Which one of the following examples maintains the lowest total collective dose?

- A. One individual performing the job in a 60 mrem/hr field for 60 minutes.
- B. One individual installing temporary shielding in a 60 mrem/hr field for 30 minutes, performing the job in a 6 mrem/hr field for 60 minutes, and then removing the temporary shielding in a 6 mrem/hr field for 20 minutes.
- C. Two individuals performing the job in a 60 mrem/hr field for 35 minutes.
- D. Two individuals installing temporary shielding in a 60 mrem/hr field for 15 minutes, performing the job in a 6 mrem/hr field for 40 minutes, and then removing the temporary shielding in a 6 mrem/hr field for 10 minutes.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 67

Immediately following a reactor scram, the following plant conditions exist:

- Reactor power 0%
- RPV pressure 940 psig
- RPV water level +196 inches
- Two control rods are stuck Position 48
- Drywell pressure 0.5 psig
- Drywell temperature 130 °F
- Containment temperature 97 °F

Which of the following Plant Emergency Instruction(s) is/are required to be entered based on the plant conditions described above?

- A. PEI-T23, Containment Control and PEI-B13, RPV Control (Non-ATWS).
- B. PEI-T23, Containment Control and PEI-B13, RPV Control (ATWS).
- C. Only PEI-B13, RPV Control (ATWS).
- D. Only PEI-T23, Containment Control.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 68

The following plant conditions exist:

- A small-line break LOCA is in progress
- The reactor has been shutdown
- Drywell temperature is 250 °F and slowly increasing
- Drywell pressure is 2.0 psig and slowly increasing

Which one of the following PEI actions would not mitigate the increase in Drywell temperature and pressure, by itself?

- A. Restore NCC to Drywell coolers.
- B. Operate all available Drywell cooling.
- C. Perform a controlled depressurization of the RPV.
- D. Irrespective of cooldown rate, depressurize the RPV rapidly using the Main Turbine Bypass Valves.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 69

The plant is operating at 50% power.

Both Reactor Recirculation Pumps trip to OFF resulting in a core flow of 30 Mlbm/hr.

Which one of the following conditions will require the Control Room Operator to immediately insert a manual reactor scram in accordance with ONI-C51, Unplanned Change in Reactor Power or Reactivity?

- A. APRM oscillations of 5% peak-to-peak are observed.
- B. The pre-transient load line was 102%.
- C. All OPRMs are currently inoperable.
- D. As directed by Reactor Engineering.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 70

A Loss of Coolant Accident has occurred.

The Safety Parameter Display System (SPDS) is unavailable.

The following plant conditions exist:

- Containment temperature            175 °F
- Drywell temperature                260 °F

Which one of the following is the lowest valid RPV water level that can be read on the Wide Range water level instruments?

**PEI-SPI Figure 2a is provided for reference.**

- A.            8 inches
- B.            11 inches
- C.            15 inches
- D.            23 inches



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 71

The plant is operating in MODE 1.

In accordance with Technical Specifications, which one of the following is the minimum Suppression Pool temperature, that if exceeded, requires the Control Room Operator to place the Reactor Mode Switch in the SHUTDOWN position?

- A. 95 °F
- B. 105 °F
- C. 110 °F
- D. 120 °F

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 72

The plant is operating at 20% power when a control rod drop event occurs.

Which one of the following alarms could be indicative of this control rod drop event?

- A.            ROD BLOCK SRM UPSC/INOP
- B.            ROD BLOCK IRM UPSCALE
- C.            ROD OVERTRAVEL
- D.            LPRM UPSCALE

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 73

During refueling activities, a loss of upper Containment pool water level will require the suspension of Core Alterations and movement of irradiated fuel after placing them in a safe condition.

Which one of the following is a 'safe' condition for an irradiated fuel bundle in the Inclined Fuel Transfer System carriage (IFTS) carriage?

The fuel bundle is properly seated in the IFTS carriage with the carriage \_\_\_\_\_.

- A. at the Raise Low Limit Carrier position with the Upender inclined.
- B. at the Bottom Out Carrier position with the Upender vertical.
- C. at the Raise Slow Carrier position with the Upender vertical.
- D. at the Fill / Drain Carrier position with the Upender inclined.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 74

The following plant conditions exist:

- The reactor scrammed on high reactor pressure
- MSIVs are isolated
- RPV water level band 192 to 200 inches
- RPV pressure band 800 to 900 psig
- Suppression Pool temperature 105 °F (increasing)
- Suppression Pool level 19.5 ft (increasing)

Which one of the following actions would improve the margin to the Heat Capacity Limit (HCL)?

**PEI-SPI Figure 4 is provided for reference.**

- A. Lower the Suppression Pool water level band to 17.8 – 18.5 feet.
- B. Lower the Suppression Pool temperature band to 90 – 95 °F.
- C. Raise the RPV water level band to 210 – 210 inches.
- D. Raise the RPV pressure band to 900 – 1000 psig.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 75**

Which one of the following lists the order of preference for indications to be used when determining Suppression Pool water temperature in accordance with the Plant Emergency Instructions?

**Note: Order of preference is defined as most preferred to least preferred.**

- A. Validated SPDS, highest reading functional instrument, Post Accident recorders.
- B. Post Accident recorders, highest reading functional instrument, validated SPDS.
- C. Highest reading functional instrument, validated SPDS, Post Accident recorders.
- D. Validated SPDS, Post Accident recorders, highest reading functional instrument.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 76

A loss of Main Condenser vacuum event is in progress.

Main Condenser pressure is 6.0 inches HgA.

Which one of the following automatic actions would occur to prevent a trip of the Main Turbine?

- A. A Load Set Runback when less than three Circulating Water Pumps are in operation.
- B. A Load Set Runback when Main Turbine Bypass Valve #1 is 50% open.
- C. A Load Limit Setback when Main Turbine Bypass Valve #1 is 50% open.
- D. A Load Limit Setback when less than three Circulating Water Pumps are in operation.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 77**

The following plant conditions exist:

- The reactor has been shutdown after a 700-day consecutive run
- The plant has been in MODE 4 for 2 days
- RHR Loop 'A' is the Shutdown Cooling mode
- Reactor Recirculation Pump 'A' is in operation
- RPV water level band is 192 to 200 inches

If RHR Pump 'A' tripped OFF due to an overcurrent fault, which one of the following describes the potential plant impact?

- A. Loss of coolant temperature monitoring.
- B. Loss of coolant conductivity monitoring.
- C. Loss of fuel cladding integrity.
- D. Loss of coolant circulation within the core.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 78

A Control Room Operator was performing rod position indication data substitution in accordance with SOI-C11 (RCIS).

When the ENT SUBST pushbutton was depressed, the red SUBST POSITION ERROR status light energized.

Which one of the following describes a potential cause for the SUBST POSITION ERROR light?

The Control Room Operator attempted to \_\_\_\_\_.

- A. replace 'bad' data from a channel containing 'good' data.
- B. replace 'bad' data from a channel containing 'substitute' data.
- C. replace 'bad' data in the same channel which already contained 'substitute' data at a different position for the same rod.
- D. replace 'bad' data in the same channel which already contained 'substitute' data at a different position for another rod.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 79

A plant startup is in progress.

Recirc Pump 'B' is running in slow speed with its flow control valve at 100% open. The Operator at the Controls has just shifted Recirc Pump 'A' to fast speed.

The following indications were received:

- Reactor power increased and stabilized at 34% power
- Reactor water level decreased to +190 inches and then was restored to normal
- RCIRC A FCV RUNBACK alarm was received on panel H13-P680

Which one of the following describes the response of the Reactor Recirculation System Flow Control Valves (FCVs)?

- A. FCVs 'A' and 'B' will remain at their present positions and only FCV 'A' runback logic has actuated.
- B. FCVs 'A' and 'B' will remain at their present positions and both FCVs 'A' and 'B' runback logic has actuated.
- C. FCV 'A' remains at its present position and FCV 'B' will runback to approximately 17% valve position and only FCV 'B' runback logic has actuated.
- D. FCV 'A' will runback to 0% valve position and FCV 'B' will runback to approximately 17% valve position and both FCVs 'A' and 'B' runback logic has actuated.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 80

The plant is operating at 100% power.

- Alarm LPCS OUT OF SERVICE is received on panel H13-P601
- Amber matrix status light LPCS LEAK DETECTED is energized
- A LPCS line break condition has been confirmed

Which one of the following describes the location of the LPCS line break?

A LPCS line break exists \_\_\_\_\_.

- A. between the discharge of the LPCS pump and the injection valve (F005).
- B. between the injection valve (F005) and the injection check valve (F006).
- C. between the RPV and the core shroud.
- D. inside the core shroud.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 81**

Which one of the following describes the response of the HPCS Pump upon receipt of a Division 3 LOCA initiation signal?

The HPCS Pump will start \_\_\_\_\_.

- A. immediately.
- B. after a 5 second time delay.
- C. after a 10 second time delay.
- D. after a 15 second time delay.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 82**

The Standby Liquid Control System (SLC) is operating to mitigate the consequences of an ATWS event.

Which one of the following conditions will require the Control Room Operators to shutdown the SLC Pumps?

- A. Reactor power is 1%.
- B. SLC storage tank level is 190 gallons.
- C. SLC storage tank temperature is 95 °F.
- D. Boron concentration in the RPV is 1000 ppm.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 83**

The following plant conditions exist:

- The Reactor Mode Switch is in the STARTUP/STANDBY position
- All IRMs are selected to Range 2
- SRM Channel 'A' is failed downscale and bypassed

Which one of the following conditions will generate a control rod withdrawal block?

- A. SRM Channel 'A' indicates 0.5 cps with its detector fully inserted.
- B. SRM Channel 'B' indicates  $2 \times 10^2$  cps with its detector fully withdrawn.
- C. SRM Channel 'C' indicates  $7 \times 10^4$  cps with its detector partially withdrawn.
- D. SRM Channel 'D' indicates 75 cps with its detector partially withdrawn.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 84

A Loss of Coolant Accident occurs with increased leakage from the Containment into the Annulus.

- AEGTS Train 'A' is in operation with its Annulus differential pressure controller in the AUTO mode
- AEGTS Train 'B' is in Secured Status

Which one of the following describes the response of AEGTS Train 'A' in order to restore the Annulus differential pressure to its desired value?

- A. The Exhaust damper throttles open and the Recirculation damper throttles closed.
- B. The Recirculation damper throttles open and the Exhaust damper throttles closed.
- C. Only the Exhaust damper throttles open.
- D. Only the Recirculation damper throttles closed.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 85**

An ATWS is in progress.

The Control Room Operator has been directed to perform PEI-SPI 1.6, Increased Cooling Water D/P.

Which one of the following describes the operation of the CRD DRIVE PRESS CONTROL VALVE, 1C11-F003, during performance of this instruction?

- A.           The valve is closed to cause the control rods to drift in.
- B.           The valve is opened to cause the control rods to drift in.
- C.           The valve is closed to cause the control rods to insert faster when manually inserting control rods.
- D.           The valve is opened to cause the control rods to insert faster when manually inserting control rods.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 86**

An isolation signal caused RWCU SUCT FM CNTMT OTBD ISOL VALVE, 1G33-F004, to automatically close; however, RWCU SUCT FM CNTMT INBD ISOL VALVE, 1G33-F001 remained open.

Which one of the following conditions caused this isolation?

- A. SLC Pump 'B' initiation.
- B. RWCU high differential flow.
- C. NRHX outlet high temperature.
- D. RWCU Pump Room 'B' high ambient temperature.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 87

The following plant conditions exist:

- The plant is in MODE 4
- RHR Loop 'A' is operating in the shutdown cooling mode using the normal return path

Which one of the following describes why RHR Pump 'A' flow must be maintained greater than 2000 gpm?

- A. To prevent excessive motor current.
- B. To prevent pump damage due to runout.
- C. To prevent a loss of RPV water inventory.
- D. To prevent voiding in the high point of the system.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 88**

During a reactor startup, a complete loss of argon gas pressure occurs in the fission chamber detector for IRM 'A'.

Which one of the following describes the IRM 'A' indication on panel H13-P680?

IRM 'A' indication will fail \_\_\_\_\_.

- A. as-is.
- B. downscale.
- C. mid-scale.
- D. upscale.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 89

An ATWS is in progress.

Emergency Depressurization is required.

The Operator at the Controls has been directed to terminate and prevent Feedwater injection in accordance with PEI-SPI 5.3, Feedwater Injection Prevention.

Which one of the following describes an improper method of Feedwater injection prevention?

- A. Shutdown all Reactor Feed Booster Pumps (RFBPs).
- B. Close the Feedwater Header Shutoff valves, 1B21-F065A & B.
- C. Place all Feedwater Flow Controllers (C34) in MANUAL with minimum (0%) output.
- D. Trip both Reactor Feed Pump Turbines (RFPTs), close the Reactor Feed Pump (RFP) Discharge Valves and Motor Feed Pump (MFP) Flow Control Valves, place the Startup Level Controller in MANUAL with minimum output, and close the FDW Pumps Bypass Valve (N27-F200).

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 90

The following plant conditions exist:

- A plant startup is in progress
- Steam Jet Air Ejector (SJAE) 'A' is in operation
- The Adsorber Vault Mode Select Switch is in the AUTO position on Off-Gas Panel H13-P845
- An alarm is received on Common Process & Area Radiation Monitoring Panel H13-P604

The Control Room Operator reports that the HI alarm (amber light) is energized on Off-Gas Post-Treatment 'A' Radiation Monitor.

Which one of the following describes the automatic response of the Off-Gas System?

- A. The Off-Gas dryers are bypassed.
- B. The Off-Gas discharge header isolates.
- C. The Off-Gas loop seal drain lines isolate.
- D. The Off-Gas adsorbers are placed in service.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 91

The plant is operating at 100% power when the following alarms occur:

- SERV AIR COMP TRBL
- TBCC HX OUTLET TEMP HIGH
- NCC HX OUTLET TEMP HIGH
- RW TO ESW ISOL DISCH TUNNEL SW FLOW LOW
- SW PUMP DISCH HEADER PRESSURE LOW

In addition, the following occur:

- RWCU SUCT FM CNTMT OTBD ISOL, 1G33-F004 isolation
- Generator stator water cooling runback

Which one of the following describes the single cause for the current plant status?

- A. Loss of Service Water.
- B. Loss of Nuclear Closed Cooling Water.
- C. Loss of Service And / Or Instrument Air.
- D. Loss of Turbine Building Closed Cooling Water.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 92

The plant is operating at 100% power.

A Main Steam Line (MSL) high flow condition is sensed which exceeds the isolation setpoint for only MSL 'A'.

Which one of the following describes the response of the MSIVs?

- A.            No MSIVs isolated.
- B.            All MSIVs isolated.
- C.            Only the inboard MSIVs isolated.
- D.            Only the MSL 'A' inboard and outboard MSIVs isolated.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 93

The plant is operating at 75% power.

Which one of the following describes the effects on the Condensate System if a reduction in Circulating Water System flow occurs?

	<u>Condenser Absolute Pressure</u>	<u>Hotwell Temperature</u>
A.	Decrease	Decrease
B.	Decrease	Increase
C.	Increase	Decrease
D.	Increase	Increase

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 94

Which one of the following modes of fuel failure may have occurred if the Linear Heat Generation Rate (LHGR) thermal limit was exceeded?

- A. 1% plastic strain.
- B. Transition boiling.
- C. 17% total cladding oxidation.
- D. 2200 °F peak cladding temperature.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 95

Which one of the following is considered to be a CORE ALTERATION during refueling with the vessel head removed and fuel in the RPV?

- A. Withdrawal of a source range monitor.
- B. Insertion of a traversing in-core probe.
- C. Removal of a jet pump assembly.
- D. Removal of a control rod blade.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

**QUESTION 96**

You have prepared a Non-Intent Conditional Change to SOI-B33, Reactor Recirculation System, in order to support the current plant startup.

The on-shift Unit Supervisor has reviewed and approved the conditional change in the "Plant Management Staff" Block on the Procedure/Instruction Change (PIC) Form.

Which one of the following additional individuals must approve the conditional change before it can become effective?

**PNPP Form No. 7309 is provided for reference.**

- A. Shift Technical Advisor
- B. Operations Manager
- C. Shift Supervisor
- D. Unit Supervisor

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 97

A Control Room Operator has completed the initial placement of the Control Room tags for a Clearance associated with RHR Loop 'A'.

You have been assigned to perform the Independent Verification for the Clearance tag placement.

You discover that the control switch for RHR 'A' Suppression Pool Suction Valve is currently in the OPEN position.

The Clearance required position for the RHR 'A' Suppression Pool Suction Valve control switch is the CLOSE position.

Which one of the following describes your expected actions?

- A. Re-position the valve control switch to the correct position and then inform the Unit Supervisor.
- B. Re-position the valve control switch to the correct position and then inform the Control Room Operator who performed the initial tag placement.
- C. Stop the independent verification and then inform the Unit Supervisor.
- D. Stop the independent verification and then inform the Control Room Operator who performed the initial tag placement to re-position the valve control switch to its correct position.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 98

A Non-Licensed Operator (NLO) is being considered for a job assignment in a High Radiation Area.

The dose rate in the job area is 120 mrem/hr. The job is expected to take 45 minutes.

The following information is known about the operator:

- His age is 35 years
- His year-to-date exposure is 950 mrem
- His year-to-date exposure at other facilities is 0 mrem
- His lifetime exposure history to date is 3500 mrem

Can the operator be assigned to this job and WHY?

- A. Yes; the operator will not exceed his initial Dose Control Level.
- B. Yes; the operator will be allowed to perform the job as long as an Increased Dose Control Level Authorization is obtained before the job.
- C. No; the operator will exceed his federal occupational dose limits.
- D. No; the operator will exceed his Dose Control Level limits which are not allowed to be increased.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 99

The plant is in MODE 4.

RHR Loop 'B' is operating in the shutdown cooling mode when the RHR Pump 'B' shaft seizes.

RHR Loop 'A' is out of service for maintenance.

Which one of the following describes an alternate method of decay heat removal in accordance with ONI-E12-2, Loss of Decay Heat Removal?

- A. Operate the High Pressure Core Spray System (HPCS) to circulate reactor coolant between the Suppression Pool and the RPV via the Reactor Head Vent.
- B. Operate the Residual Heat Removal 'C' System (RHR C) to circulate reactor coolant between the Suppression Pool and the RPV via the Safety Relief Valves (SRVs).
- C. Operate the Reactor Feed Booster Pumps (RFBPs) to maintain RPV water level while dumping reactor coolant to the Main Condenser via the Main Steam Lines.
- D. Operate the Low Pressure Core Spray System (LPCS) to maintain RPV water level while dumping reactor coolant to the Main Condenser via the Reactor Water Cleanup System (RWCU) blowdown line.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
REACTOR OPERATOR**

QUESTION 100

The plant is operating at 100% power.

Just prior to shift change, the on-coming Fire Brigade Leader (FBL) calls in sick.

The Shift Supervisor is not able to obtain another FBL by shift change.

Can the off-going FBL be allowed to leave at shift change and WHY?

- A. No; the FBL position cannot be left unmanned at shift change.
- B. No; the FBL position cannot be left unmanned under any circumstances.
- C. Yes; the FBL position may be left unmanned for a period of time not to exceed 2 hours.
- D. Yes; the FBL position may be assumed by an on-coming Non-Licensed Operator who is a member of the Fire Brigade.

**U.S. Nuclear Regulatory Commission  
Site-Specific  
Written Examination**

**Applicant Information**

Name:	Region: I / II / III / IV
Date:	Facility/Unit: <b>Perry</b>
License Level: RO / <b>SRO</b>	Reactor Type: W / CE / BW / <b>GE</b>
Start Time:	Finish Time:

**Instructions**

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. The passing grade requires a final grade of at least 80.00 percent. Examination papers will be collected five hours after the examination starts.

**Applicant Certification**

All work done on this examination is my own. I have neither given nor received aid.

\_\_\_\_\_  
Applicant's Signature

**Results**

Examination Value	<u>100.00</u>	Points
Applicant's Score	_____	Points
Applicant's Grade	_____	Percent

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 1**

The plant is operating at 100% power. The following equipment is in operation:

- CRDH Pump 'A'
- NCC Pump 'A' and 'B'

A loss of power to Bus EH11 occurs when the Preferred Source Breaker trips open.

Which one of the following describes the expected response of circuit breakers associated with Bus EH11?

- A. Bus EH11 Stub Bus Breaker opens and NCC Pump 'A' breaker opens.
- B. Bus EH11 Stub Bus Breaker opens and CRDH Pump 'A' breaker remains closed.
- C. Bus EH11 Stub Bus Breaker remains closed and CRDH Pump 'A' breaker opens.
- D. Bus EH11 Stub Bus Breaker remains closed and NCC Pump 'A' breaker remains closed.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 2

Prior to a manual reactor scram per IOI-8, Shutdown By Manual Reactor Scram, the plant is operating at 100% power with the Feedwater Master Level Controller tape set at 200 inches.

Which one of the following describes the response of the Feedwater Level Control System following the manual reactor scram signal?

Assume no operator actions are taken.

- A.           Upon receipt of the manual reactor scram signal, the level demand signal will be 196 inches for 10 seconds and then decrease to 178 inches.
- B.           Upon receipt of the manual reactor scram signal, the level demand signal will be 200 inches for 10 seconds and then decrease to 178 inches.
- C.           When RPV water level decreases to 178 inches, the level demand signal will be 196 inches for 10 seconds and then decrease to 178 inches.
- D.           When RPV water level decreases to 178 inches, the level demand signal will be 200 inches for 10 seconds and then decrease to 178 inches.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 3

The plant is operating at 95% power.

- SB&PR Channel 'A' is in service
- SB&PR Channel 'B' is in TEST

The main steam pressure transmitter output signal to SB&PR Channel 'A' fails upscale.

Which one of the following describes the valve response associated with the SB&PR System?

Assume no operator actions are taken.

- A. Turbine Control Valves and Bypass Valves open.
- B. Turbine Control Valves and Bypass Valves remain 'as-is'.
- C. Turbine Control Valves remain 'as-is' and Bypass Valves open.
- D. Turbine Control Valves open and Bypass Valves remain closed.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 4**

Which one of the following correctly lists the signal(s) which will automatically close the Containment isolation valves for the Drywell Equipment and Floor Drain Sump Systems?

- A. High Drywell pressure (1.68 psig) or low Reactor water level (L1).
- B. High Drywell pressure (1.68 psig) or low Reactor water level (L2).
- C. Drain sump high discharge temperature.
- D. Drain sump pump high discharge pressure.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 5

An ATWS is in progress.

PEI-B13, RPV Control (ATWS) directs the Control Room Operators to “Stabilize RPV pressure to less than 1000 psig”.

Which one of the following conditions meets the intent of this PEI action step?

- A. Automatic operation of the Bypass Valves.
- B. Automatic operation of the SRVs on Low-Low Setpoint.
- C. Uncontrolled depressurization due to a stuck-open Bypass Valve.
- D. Manual operation of SRVs when sufficient Bypass Valve capacity exists.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 6

During an incomplete scram, which one of the following Redundant Reactivity Control System (RRCS) control signals requires an 'APRM Not Downscale' permissive?

- A. SLC Pump trip
- B. Feedwater Runback (FWRB)
- C. Alternate Rod Insertion (ARI)
- D. Recirc Pump Transfer to LFMG

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 7

ONI-C61, Evacuation of the Control Room, has been entered.

The reactor was not scrammed prior to leaving the Control Room.

Which one of the following is the preferred method to fully insert all control rods from outside the Control Room?

- A. Open the specified scram air header drain valves.
- B. Open the specified RPS MG set output breakers.
- C. Cycle the specified RPS power distribution panel breakers.
- D. Cycle the specified ATWS UPS distribution panel breakers.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 8

The FHB Ventilation System is in operation in accordance with SOI-M40. During movement of irradiated fuel in the FHB, an irradiated fuel bundle is dropped.

Shortly thereafter, a HIGH radiation alarm is received on the FHB Ventilation Exhaust GAS module.

Which one of the following describes the current status of the FHB Ventilation System?

- A.            No Exhaust Fan running; no Supply Fan running.
- B.            No Exhaust Fan running; one Supply Fan running.
- C.            Two Exhaust Fans running; no Supply Fan running.
- D.            Two Exhaust Fans running; one Supply Fan running.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 9

The plant has experienced a Loss of Coolant Accident due to a complete break of the Recirculation System piping.

Which one of the following describes the effects of Drywell pressure?

Drywell pressure will rise to a maximum value, thereby \_\_\_\_\_.

- A. clearing the Drywell to Containment Horizontal Vents, releasing steam directly into the Containment and pressurizing Containment to a maximum value.
- B. clearing the Drywell to Containment Horizontal Vents and causing a rise in Containment pressure followed by a lowering of Drywell pressure and recovering of the vents.
- C. covering the Drywell to Containment Horizontal Vents and preventing a rise in Containment pressure.
- D. preventing the uncovering of the Drywell to Containment Horizontal Vents and preventing a rise in Containment pressure.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 10

Which one of the following parameters will require the initiation of a reactor scram in accordance with the Plant Emergency Instructions (PEIs)?

Assume that the reactor is operating and no ATWS condition exists.

- A. Drywell pressure is 1.2 psig.
- B. Suppression Pool level is 14.50 ft.
- C. Containment temperature is 185°F.
- D. Suppression Pool temperature is 97°F.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 11**

Why is a level of 7.25 feet in the Suppression Pool a concern to the Control Room Operators when operating in the PEIs?

- A.            Operation at this level could cause air entrapment at the RCIC suction strainer.
- B.            Operation at this level will uncover the Suppression Pool suction strainer.
- C.            Operation at this level could result in exceeding the stress limits of the SRV tail pipe.
- D.            Operation at this level will cause rapid pressurization of Containment during an SRV lift.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 12

The plant is operating at 50% power with both Reactor Feed Pump Turbines (RFPTs) on the Master Level Controller (MLC).

Which one of the following describes the expected response of the Feedwater System if the Narrow Range level instrument input to the MLC fails upscale?

Assume no operator actions are taken?

- A. Feedwater flow increases; both RFPTs trip on mechanical overspeed.
- B. Feedwater flow increases; both RFPTs trip on high RPV level (L8).
- C. Feedwater flow decreases; both RFPTs trip on low RPV level (L2).
- D. Feedwater flow decreases; both RFPTs trip on RCIC initiation.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 13

Which one of the following describes the reactor core conditions that Cold Shutdown Boron Weight is based upon in order to provide an adequate shutdown margin?

- A. 70°F fuel/moderator temperature, xenon free, and all control rods fully withdrawn.
- B. 70°F fuel/moderator temperature, xenon free, and 50% rod density.
- C. 68°F fuel/moderator temperature, xenon free, and all control rods fully withdrawn.
- D. 68°F fuel/moderator temperature, xenon free, and 50% rod density.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 14

PEI-D17, Radioactivity Release Control contains the following action step:

“Except for systems required to assure adequate core cooling or shutdown the reactor, isolate all primary systems that are discharging into areas outside one or more of the following: Annulus, Auxiliary Building, Intermediate Building, or Steam Tunnel.”

Which one of the following defines the term “primary system” as used in PEI-D17?

A “primary system” refers to any system \_\_\_\_\_.

- A. that can be used to reduce RPV pressure.
- B. that can be used to maintain RPV water level.
- C. connected to the RPV and contains radioactive coolant.
- D. connected to the RPV and will have a reduced leak rate if RPV pressure is lowered.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 15

The following plant conditions exist:

- A Loss of Coolant Accident has occurred
- Hydrogen is present in the Primary Containment
- PEI-M51/56, Hydrogen Control, has been entered
- Hydrogen Recombiners have been started

Which one of the following hydrogen concentrations will require the Hydrogen Recombiners to be secured, including the bases for this action?

The Hydrogen Recombiners are secured at \_\_\_\_\_.

- A. 4% hydrogen concentration in order to prevent their becoming an ignition source.
- B. 4% hydrogen concentration because there is insufficient oxygen available to support the recombination reaction.
- C. 6% hydrogen concentration because there is insufficient oxygen available to support the recombination reaction.
- D. 6% hydrogen concentration in order to prevent their becoming an ignition source.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 16

The plant is operating at 100% power. Reactor Narrow Range Level Channel 'B' is selected for input to the Master Level Controller. DC Bus D-1-A de-energizes due to a fault condition.

Which one of the following describes the consequences of the failure to perform the immediate actions of ONI-R42-4, Loss of DC Bus D-1-A?

- A.           RPV water level will increase causing a reactor scram but RFPT 'A' and the Main Turbine will not automatically trip on high water level.
- B.           RPV water level will increase until the RFPTs trip on high water level and the reactor will scram when water level decreases to RPV Level 3.
- C.           RPV water level will increase until the RFPTs trip on high water level but a reactor scram will not occur due to the loss of DC Bus D-1-A.
- D.           RPV water level will increase and then stabilize at a higher level when the level error signal overcomes the flow error signal to RFPT 'A'.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 17

Which one of the following describes the effect on the Reactor Recirculation System for an actuation of the End of Cycle Recirculation Pump Trip (EOC-RPT) logic following a Main Turbine trip?

- A. At greater than 38% rated thermal power, only the CB5 breaker will trip, the Low Frequency Motor Generator (LFMG) will start, and the CB2 breaker will close for each Recirculation Pump.
- B. At greater than 38% rated thermal power, the CB3, CB4, and CB5 breakers will trip, the Low Frequency Motor Generator (LFMG) will start, and the CB2 breaker will close for each Recirculation Pump.
- C. At less than 38% rated thermal power, only the CB5 breaker will trip, the Low Frequency Motor Generator (LFMG) will start, and the CB2 breaker will close for each Recirculation Pump.
- D. At less than 38% rated thermal power, the CB2, CB3, and CB4 breakers will trip for each Recirculation Pump.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 18

The High RPV Water Level Trip (L8) is designed to prevent \_\_\_\_\_.

- A. RCIC Turbine damage due to water slugging.
- B. MSL hanger damage due to excessive weight.
- C. Main Turbine damage due to moisture carryover.
- D. MSIV damage due to excessive hydraulic loading.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 19**

Which one of the following describes the bases for maximizing Containment cooling during the execution of PEI-T23, Containment Control?

- A. To prevent exceeding the Containment design temperature limit of 330°F.
- B. To prevent exceeding the Containment average air temperature LCO limit of 145°F.
- C. To prevent exceeding the environmental qualification temperature of 330°F for safety-related electrical equipment in the Containment.
- D. To prevent exceeding the environmental qualification temperature of 185°F for safety-related electrical equipment in the Containment.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 20

The plant is operating at 100% power.

- TBCC HX OUTLET TEMP HIGH alarm is received on panel H13-P870
- ONI-P44, Loss of Turbine Building Closed Cooling, has been entered
- TBCC Heat Exchanger Outlet Temperature Control Valve, 1P41-F003, was confirmed to have failed in the 'close' position

Which one of the following describes the correct action to perform in accordance with ONI-P44?

- A. Contact I&C to troubleshoot the temperature controller for 1P41-F003.
- B. Direct a Non-Licensed Operator to manually throttle open 1P41-F003.
- C. Open TBCC Heat Exchanger Service Water Temperature Control Valve Bypass Valve, 1P41-F390.
- D. Shutdown a Service Water Pump per SOI-P40/41 until only one Service Water Pump is operating.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 21

The plant is in MODE 2 when the running CRDH Pump trips.

CRD charging water header pressure decreases to reactor pressure.

Which one of the following describes the plant conditions that would require immediately placing the Reactor Mode Switch in the SHUTDOWN position in accordance with ONI-C11-1, Inability To Move Control Rods?

- A.           Reactor pressure is 500 psig.  
              Accumulator fault occurs on control rod 20-27 at position 00.
- B.           Reactor pressure is 500 psig.  
              Accumulator fault occurs on control rod 20-27 at position 24.
- C.           Reactor pressure is 700 psig.  
              Accumulator fault occurs on control rod 20-27 at position 00.
- D.           Reactor pressure is 700 psig.  
              Accumulator fault occurs on control rod 20-27 at position 24.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 22

Which one of the following Plant Emergency Instruction (PEI) curves or limits, if exceeded, could directly result in a loss of Primary Containment integrity?

- A. Primary Containment Limit (PCL).
- B. Pressure Suppression Pressure (PSP).
- C. SRV Tail Pipe Level Limit (SRVTPLL).
- D. Maximum Core Uncovery Time Limit (MCUTL).

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 23

Which one of the following describes the bases for isolating any system discharging into the annulus and surrounding containment in accordance with PEI-N11, Containment Leakage Control?

- A. To terminate rising temperatures, radiation levels, and water levels in the Secondary Containment.
- B. To protect equipment in the Annulus and Primary Containment.
- C. To minimize reactor coolant inventory loss.
- D. To preserve Turbine Building accessibility.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 24**

Which one of the following alarms, if received on AB EL 574' EAST area radiation monitor, would require entry into PEI-N11, Containment Leakage Control?

- A. Fail
- B. Alert
- C. High
- D. High-High

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 25

The following plant conditions exist:

- A steam leak has occurred in the RCIC Pump Room
- HIGH radiation alarm is received on the Auxiliary Building Ventilation Exhaust GAS module
- Auxiliary Building Ventilation Exhaust GAS module indicates 29,000 cpm
- PEI-N11, Containment Leakage Control, has been entered

Which one of the following describes the correct action to perform in accordance with PEI-N11?

**PEI-N11 flowchart is provided for reference.**

Direct a Non-Licensed Operator to \_\_\_\_\_.

- A. start a second Auxiliary Building Ventilation Supply Fan.
- B. verify the Auxiliary Building Ventilation Supply Fan is tripped.
- C. secure the Auxiliary Building Ventilation Supply and Exhaust Fans.
- D. verify the Auxiliary Building Ventilation Supply and Exhaust Fans are running.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 26

The following plant conditions exist:

- A 25 gpm leak has occurred on the ESW-side of the RHR 'A' Heat Exchanger
- RHR A PUMP ROOM SUMP LEVEL HIGH alarm is received on panel H13-P601

Which one of the following describes how this leak is captured by the Liquid Radwaste Sumps System (G61)?

Assume no operator actions have been taken.

- A. The RHR 'A' Pump Room sump drain valve will automatically open to route the leakage to the Auxiliary Building Floor Drain Sump.
- B. The RHR 'A' Pump Room sump drain valve will automatically close to contain the leakage to the RHR 'A' Pump Room.
- C. The RHR 'A' Pump Room sump will gravity drain to the Auxiliary Building Floor Drain Sump.
- D. The RHR 'A' Pump Room sump will fill and overflow into the RHR 'A' Pump Room.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 27**

During a reactor startup, control rods are being withdrawn to achieve 50% rod density.

- Rod Sequence 'A' is selected
- All Group 5 through 10 control rods are fully inserted
- All Group 1 and 2 control rods have been fully withdrawn
- All Group 3 control rods have been withdrawn to notch position '12'

The Control Room Operator selects a Group 4 control rod for withdrawal.

Which one of the following describes the response of the Rod Control and Information System (RC&IS)?

- A. Withdrawal of the control rod is immediately blocked.
- B. Withdrawal of the control rod will occur with no restrictions.
- C. Withdrawal of the control rod will be blocked when the control rod reaches notch position '04'.
- D. Withdrawal of the control rod will be blocked when the control rod reaches notch position '12'.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 28

The plant is operating at 100% power.

- AFDL IN CONTROL alarm is received on panel H13-P680
- APRM Channel 'A' failed upscale
- Fifteen (15) seconds later, APRM Channel 'A' is inadvertently bypassed

Assume no additional operator actions are taken.

Which one of the following describes the response of the Reactor Recirculation System Flow Control Valves (FCVs)?

- A.           The Reactor Recirculation Loop Flow Controllers will return the FCVs to their pre-transient valve positions.
- B.           The Reactor Recirculation Loop Flow Controllers will maintain the FCVs at their current valve positions.
- C.           The Automatic Flow Demand Limiter will return the FCVs to their pre-transient valve positions.
- D.           The Automatic Flow Demand Limiter will maintain the FCVs at their current valve positions.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 29

Which one of the following describes the operation of the RHR PUMP 'A' & HX COOLER, M39-B001A?

The RHR PUMP 'A' & HX COOLER will automatically start \_\_\_\_\_.

- A.       when an RHR LOCA initiation signal seals-in (K110A relay contact); room heat is dissipated by recirculating room air through a heat exchanger cooled directly by ECC.
- B.       when an RHR LOCA initiation signal seals-in (K110A relay contact); room heat is dissipated by recirculating room air through a heat exchanger cooled directly by ESW.
- C.       when RHR Pump 'A' breaker closes (52a contact); room heat is dissipated by recirculating room air through a heat exchanger cooled directly by ECC.
- D.       when RHR Pump 'A' breaker closes (52a contact); room heat is dissipated by recirculating room air through a heat exchanger cooled directly by ESW.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 30

One hour ago the Control Room Operators discovered that the blue pressure permissive light for the LPCS Injection Valve, 1E21-F005, was not lit.

Control Room Operators confirmed the blue light bulb was good.

Which one of the following describes the operation of the LPCS Injection Valve control logic if a Loss-of Coolant Accident occurs?

- A. The LPCS Injection Valve automatically opens, irrespective of RPV pressure, due to the LPCS LOCA initiation signal.
- B. The LPCS Injection Valve remains closed and cannot be opened with its control switch until RPV pressure decreases to 600 psig.
- C. The LPCS Injection Valve remains closed and cannot automatically open until RPV pressure decreases to 530 psig.
- D. The LPCS Injection Valve remains closed and cannot automatically open until RPV pressure decreases to 600 psig.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 31

The following plant conditions exist:

- A Loss of Coolant Accident has occurred
- Drywell pressure is 1.8 psig
- RPV water level is +195 inches and stable
- The High Pressure Core Spray (HPCS) Pump has been overridden to STOP

Subsequently, Bus EH13 loses power and is re-energized by the HPCS Diesel Generator.

Assume no additional operator actions were taken.

Which one of the following describes the current condition of the HPCS Pump?

The HPCS Pump is \_\_\_\_\_.

- A. not running because the initiation logic was reset.
- B. not running because the override logic was not affected.
- C. running because the override logic was reset.
- D. running because the initiation logic was not affected.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 32

The following plant conditions exist:

- MODE 5
- RPS INST VOL HI annunciator on panel H13-P680 is in alarm
- All INST VOL LEVEL HI SCRAM BYPASS switches are in BYPASS
- RPS logic is reset

Which one of the following describes the effect on the Reactor Protection System (RPS) when the Reactor Mode Switch is taken from SHUTDOWN, through REFUEL, to the STARTUP/STANDBY position?

- A. RPS actuation occurs because the Scram Discharge Volume High trip is enabled when the Reactor Mode Switch is in the STARTUP/STANDBY position.
- B. RPS actuation occurs because the Scram Discharge Volume High trip is enabled when the Reactor Mode Switch is in the REFUEL position.
- C. RPS actuation does not occur because the Scram Discharge Volume High trip is bypassed when the Reactor Mode Switch is in the STARTUP/STANDBY position.
- D. RPS actuation does not occur because the Scram Discharge Volume High trip is bypassed when the INST VOL LEVEL HI SCRAM BYPASS switches are in BYPASS.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 33

The plant is operating at 75% power.

Which one of the following setpoints is displayed when the Operator at the Controls depresses the APRM ALARM LEVEL RECORD pushbutton for an IRM/APRM Recorder?

- A. Fixed neutron flux upscale rod block
- B. Fixed neutron flux upscale scram
- C. Flow biased rod block
- D. Flow biased scram



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 34

RHR Loop 'B' is operating in the shutdown cooling mode when a loss of RPS Bus 'A' occurs.

Which one of the following describes the response of RHR Loop 'B'?

- A. RHR Loop 'B' remains in operation.
- B. Only an RHR SDC Inboard isolation occurs.
- C. Only an RHR SDC Outboard isolation occurs.
- D. An RHR SDC Inboard and Outboard isolation occurs.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 35

The following plant conditions exist:

- A Station Blackout (SBO) is in progress
- RCIC System is maintaining RPV water level
- RCIC suction is from the CST

Which one of the following describes the response of the RCIC suction valve(s) if a high Suppression Pool (SP) level occurs?

- A. The RCIC suction valves fail 'as-is'.
- B. The SP suction valve remains closed.
- C. The CST suction valve remains open.
- D. The RCIC suction valve transfer occurs.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 36

Which one of the following is the power source to the ADS 'B' solenoid valves?

- A. ED-1-A
- B. ED-1-B
- C. D-1-A
- D. D-1-B

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 37**

The following plant conditions exist:

- A Loss of Coolant Accident has occurred
- Hydrogen is present in the Drywell
- PEI-M51/56, Hydrogen Control, has been entered
- Combustible Gas Mixing Compressors have been started

Which one of the following conditions would cause Combustible Gas Mixing Compressor 'A' to trip?

- A. RHR Pump 'A' control switch taken to STOP.
- B. Less than 17 gpm cooling water flow to the oil cooler.
- C. Less than 80 gpm cooling water flow to the air aftercooler.
- D. LPCI A Injection Valve, 1E12-F042A, less than 90% open.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 38

The following plant conditions exist:

- A Loss of Coolant Accident has occurred
- PEI-M51/56, Hydrogen Control, has been entered
- Hydrogen Igniters were started
- Containment hydrogen concentration has exceeded HDOL

Which one of the following conditions would require the Hydrogen Igniter control switches to be placed in the OFF position?

- A. A loss of power occurs to the Hydrogen Analyzers.
- B. A loss of power occurs to the Hydrogen Igniters.
- C. An oxygen-inert environment exists in Containment.
- D. A steam-inert environment exists in Containment.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 39

Following a reactor scram, RPV water level decreased to +100 inches.

Which one of the following systems will have automatically isolated?

- A. Nuclear Closed Cooling Water System (P43)
- B. Safety-Related Instrument Air System (P57)
- C. Reactor Water Cleanup System (G33)
- D. Fire Service Water System (P54WTR)

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 40

The following plant conditions exist:

- RHR loop 'A' started in the LPCI mode on a high Drywell pressure signal
- RHR Pump 'A' was shutdown by taking its control switch to the STOP position

Which one of the following conditions will automatically re-start RHR Pump 'A'?

Assume no further operator actions are taken.

- A. Automatic Depressurization System (ADS) initiation signal occurs.
- B. Drywell pressure signal clears and re-occurs.
- C. Containment Spray initiation signal occurs.
- D. RPV low water level (L1) signal occurs.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 41

An SRV tailpipe vacuum breaker has failed in the 'open' position during SRV operations.

Which one of the following conditions will result, potentially requiring entry into PEI-T23, Containment Control?

- A. Water will be drawn up into the tailpipe from the Suppression Pool each time an SRV is closed.
- B. Steam will bypass the quencher and discharge directly into the Suppression Pool each time an SRV is opened.
- C. Steam will discharge from the tailpipe directly into the Containment airspace each time an SRV is opened.
- D. Steam will discharge from the tailpipe directly into the Drywell airspace each time an SRV is opened.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 42

The following plant conditions exist:

- Main Turbine roll is in progress
- STARTING RATE – MEDIUM has been selected
- SPEED SET RPM – 1800 has been selected
- Main Turbine speed is 500 rpm and increasing

Which one of the following is the effect on the acceleration rate of the Main Turbine if one of the acceleration input signals in the Speed Control Unit is lost?

The acceleration rate will \_\_\_\_\_.

- A. decrease by 1/3 (one-third).
- B. decrease by 1/2 (one-half).
- C. remain the same.
- D. double.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 43**

Which one of the following sets of Feedwater controls provides a speed demand signal to the Reactor Feed Pump Turbines?

**Note: A partial list will be incorrect.**

- A. Manual Speed Control Dial, Startup Level Controller, and Master Level Controller.
- B. Low Flow Controller, Manual Speed Control Dial, and Startup Level Controller.
- C. Startup Level Controller, Low Flow Controller, and Master Level Controller.
- D. Master Level Controller, Manual Speed Control Dial, and Low Flow Controller.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 44

The plant is operating at 40% power.

- REACTOR NARROW RANGE LEVEL Switch, 1C34-S1, is selected to Channel 'A'
- Reactor Narrow Range Channel 'A' indicates +196 inches on 1C34-R606A
- Reactor Narrow Range Channel 'B' indicates +193 inches on 1C34-R606B
- The Master Level Controller is in operation with its tapeset at +196 inches

Which one of the following describes the RPV water level response, as indicated on Reactor Narrow Range Channel 'B', when the Control Room Operator switches 1C34-S1 to Channel 'B'?

RPV water level initially \_\_\_\_\_.

- A. decreases and then stabilizes at +193 inches.
- B. decreases and then stabilizes at +196 inches.
- C. increases and then stabilizes at +196 inches.
- D. increases and then stabilizes at +199 inches.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 45

A Main Turbine/Generator trip occurred and station loads failed to transfer to the Unit 1 Startup Transformer.

Which one of the following Immediate Actions is required by ONI-N32, Turbine And/Or Generator Trip?

- A. Close START-UP SUPPLY BRKRS L1006 and L1009.
- B. Open NORMAL SUPPLY BRKRS, L1102 and L1202.
- C. Enter ONI-R10, Loss Of AC Power.
- D. Enter ONI-R22-1, Loss Of An Essential And / Or A Stub 4.16 KV Bus.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 46**

A Loss of Off-Site Power (LOOP) occurs.

RPV water level decreases to +10 inches.

Ten minutes later, the Division 1 DG jacket water pump fails.

Which one of the following describes the effect on Bus EH11 due to the Division 1 DG jacket water pump failure?

- A. Bus EH11 de-energizes because the Division 1 DG trips on high jacket water temperature.
- B. Bus EH11 de-energizes because the Division 1 DG trips on high lube oil temperature.
- C. Bus EH11 remains energized because the Division 1 DG high jacket water temperature trip is bypassed on a LOOP signal.
- D. Bus EH11 remains energized because the Division 1 DG high jacket water temperature trip is bypassed on a LOCA signal.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 47

If a Reactor Recirculation Pump is started at a reduced RPV water level, then RPV water level in the downcomer annulus will initially \_\_\_\_\_.

- A. decrease which may cause an RPS actuation due to RPV Level 3.
- B. decrease which may cause cavitation of the Flow Control Valves.
- C. increase which may cause an RPS actuation due to high neutron flux.
- D. increase which may cause an excessive cooldown of the RPV.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 48

Which one of the following requirements provides for adequate NPSH for RHR Pump 'A' during startup or operation in the Suppression Pool Cooling mode?

- A. The LPCS & RHR A waterleg pump must be running before RHR Pump 'A' will start.
- B. The Suppression Pool suction valve must be open before RHR Pump 'A' will start.
- C. The LPCS System must not be operated in the LPCS Test mode during RHR 'A' Suppression Pool Cooling mode operation.
- D. The RHR System flow must be maintained greater than 6000 gpm during RHR 'A' Suppression Pool Cooling mode operation.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 49**

A refueling outage is in progress. One control rod is selected and withdrawn with the Reactor Mode Switch in the REFUEL position.

Which one of the following Refueling Platform (F15) operations would be prevented?

- A. Removing a fuel assembly from the RPV with the Main Fuel Hoist.
- B. Removing a control rod blade from the RPV with the Auxiliary Hoist.
- C. Moving the Refueling Platform inside the RPV with the Main Fuel Hoist unloaded.
- D. Moving the Refueling Platform from IFTS to the RPV with the Main Fuel Hoist unloaded.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 50

The plant is operating at 100% power with the following Main Generator conditions:

- Electrical output            1000 MWe
- Hydrogen pressure        60 psig
- Power factor                1

Select the maximum allowable reactive load for the Main Generator under these conditions.

**PDB-C0002, Generator Capability Curve is provided for reference.**

- A.            575 MVars
- B.            675 MVars
- C.            700 MVars
- D.            775 MVars

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 51

Which one of the following conditions will cause the static transfer switch in the Plant Vital Balance of Plant uninterruptible power supply (BOP-UPS) system to automatically shift?

- A. Low cooling air flow to the inverter.
- B. High voltage sensed at the output of the inverter.
- C. Ground fault sensed on Vital Distribution Bus V-1-A.
- D. Loss of battery chargers to Battery D-1-A for more than 15 minutes.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 52

DC control power is lost to Bus H11.

Which one of the following describes the operation of circuit breakers supplying loads from Bus H11?

The circuit breakers can \_\_\_\_\_.

- A. be opened and closed locally at the breaker cubicle.
- B. be closed from the Control Room and opened locally at the breaker cubicle.
- C. be closed locally at the breaker cubicle, however, all circuit breaker automatic trip functions are still available.
- D. be opened and closed from the Control Room, however, all circuit breaker automatic trip functions are disabled.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 53

An air purge of the Off-Gas System has been established in accordance with SOI-N64/62, Off-Gas / Condenser Air Removal System, during a plant startup.

Which one of the following describes the source of purge air and the point of injection into the Off-Gas System?

- A. Instrument Air enters at the inlet to the recombiners.
- B. Instrument Air enters at the inlet to the adsorbers.
- C. Service Air enters at the inlet to the preheaters.
- D. Service Air enters at the inlet to the gas dryers.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 54

The Fire Service Water System is in Standby Readiness when a fire system initiation occurs. Fire main pressure decreases to 110 psig and then increases to 150 psig.

Which one of the following describes the current configuration of the Fire Service Water System?

Assume no operator actions were performed.

Fire Service Jockey Pump	Motor Fire Service Pump	Diesel Fire Service Pump
-----------------------------	----------------------------	-----------------------------

- |    |         |         |         |
|----|---------|---------|---------|
| A. | Running | Off     | Off     |
| B. | Off     | Running | Off     |
| C. | Running | Running | Running |
| D. | Off     | Off     | Running |

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 55

The Non-Licensed Operator reports that the guide vanes on the operating Control Complex Chiller (P47) have failed closed.

Which one of the following describes the potential impact of this condition?

Control Room air temperature will \_\_\_\_\_.

- A. decrease; personnel habitability limits may be exceeded.
- B. decrease; equipment temperature limits may be exceeded.
- C. increase; equipment humidity limits may be exceeded.
- D. increase; equipment temperature limits may be exceeded.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 56

A Non-Licensed Operator reports that the refrigeration unit for an in-service Instrument Air Dryer is not operating.

This condition, if left uncorrected, may result in the introduction of \_\_\_\_\_ into the Instrument Air System.

- A. foreign particles
- B. wear products
- C. hydrocarbons
- D. water

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 57

The following equipment conditions exist:

- All ECCS and RCIC are in Standby Readiness
- Division 1 and 2 Diesel Generators are in Standby Readiness
- HPCS Diesel Generator is in Secured Status

A reactor scram occurs and RPV water level decreases to +120 inches.

Two minutes later, which one of the following describes the configuration of the Emergency Service Water (ESW) System?

Assume no operator actions are performed.

	<u>ESW Pump A</u>	<u>ESW Pump C</u>
A.	Off	Off
B.	Running	Off
C.	Running	Running
D.	Off	Running



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 58

A reactor startup/heatup is in progress with IRMs on Range 7.

During a single notch withdrawal of control rod 30-31 from position 08 to 10, the collet fingers stick in the 'unlocked' position.

Which one of the following describes the effect of this control rod withdrawal event?

Control rod 30-31 will \_\_\_\_\_.

- A. settle at position 10; reactor power and heatup rate will stabilize.
- B. drift into the core; reactor power and heatup rate will decrease.
- C. drift out of the core; reactor power and heatup rate will increase.
- D. remain at position 08 until drive water header pressure has been increased sufficiently to free the collet fingers.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 59**

The Fuel Pool Cooling and Cleanup (FPCC) System is in operation when a RHR LOCA initiation occurs.

Which one of the following describes the cooling water flowpath configuration for the FPCC heat exchangers?

- A. Nuclear Closed Cooling water flowpath automatically isolates; Emergency Service Water flowpath automatically lines up to provide cooling water flow.
- B. Nuclear Closed Cooling water flowpath automatically isolates; Emergency Service Water flowpath must be manually lined up to provide cooling water flow.
- C. Nuclear Closed Cooling water flowpath must be manually isolated; Emergency Service Water flowpath must be manually lined up to provide cooling water flow.
- D. Nuclear Closed Cooling water flowpath must be manually isolated; Emergency Service Water flowpath automatically lines up to provide cooling water flow.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 60

Which one of the following Radiation Monitors will isolate the Containment Vessel and Drywell Purge System (M14) if a high radiation condition occurs?

- A. Drywell Atmosphere Radiation Monitor, 1D17-K670.
- B. Containment Atmosphere Radiation Monitor, 1D17-K680.
- C. Containment Ventilation Exhaust Radiation Monitor, 1D17-K609A-D.
- D. Containment & Drywell Purge Exhaust Radiation Monitor, 1D17-K660.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 61

Which one of the following describes the procedural requirements for performance of a Valve Lineup Instruction (VLI)?

- A.       The valves can be positioned in any order unless a specific order is specified by the Unit Supervisor.  
          The Independent Verifier can verify the valves in any order.
- B.       The valves can be positioned in any order unless a specific order is specified by the Unit Supervisor.  
          The Independent Verifier must verify the valves in the same order.
- C.       The valves shall be positioned in the order in which they appear unless specified otherwise by the Unit Supervisor.  
          The Independent Verifier must verify the valves in the same order.
- D.       The valves shall be positioned in the order in which they appear unless specified otherwise by the Unit Supervisor.  
          The Independent Verifier can verify the valves in any order.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 62

The following conditions exist:

- You are an on-coming Control Room Operator (SO/US/SS)
- This is your first day back on-shift after 10 days of time off

Which one of the following describes the requirement for reviewing the Daily and Standing Instructions?

The Daily and Standing Instructions should be reviewed for the previous \_\_\_\_\_.

- A. one day.
- B. two days.
- C. seven days.
- D. ten days.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 63

An Infrequently Performed Test or Evolution (IPTE) is in progress which will demonstrate the heat removal capability of the RHR heat exchangers.

The following plant conditions exist:

- Reactor power is being maintained at 5%
- RCIC is operating in the CST-to-CST mode in order to raise Suppression Pool temperature to  $100 \pm 2$  °F
- Suppression Pool temperature is 97 °F and slowly increasing

Which one of the following describes the entry requirements for PEI-T23, Containment Control, and the Required Action requirements for LCO 3.6.2.1, Suppression Pool Average Temperature?

- A. PEI-T23 is not required to be entered; the Required Actions for LCO 3.6.2.1 are not required to be performed.
- B. PEI-T23 is not required to be entered; the Required Actions for LCO 3.6.2.1 are required to be performed.
- C. PEI-T23 is required to be entered; the Required Actions for LCO 3.6.2.1 are not required to be performed.
- D. PEI-T23 is required to be entered; the Required Actions for LCO 3.6.2.1 are required to be performed.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 64

A General Emergency is in progress.

Which one of the following individuals is directly charged with the command authority over all activities involving plant operations.

- A. EOF Emergency Coordinator
- B. TSC Operations Manager
- C. Shift Supervisor
- D. Unit Supervisor

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 65

I&C has been authorized to start SVI-C71-T0042B, Drywell High Pressure Channel B Functional for 1C71-N650B.

The Operator at the Controls has been notified that the following alarms will occur during performance of this SVI:

- |                                     |             |
|-------------------------------------|-------------|
| • RPS DW PRESS HI                   | P680-05A-B8 |
| • 1/2 SCRAM B/D                     | P680-05A-B9 |
| • RPS B & D OUT OF SERVICE          | P680-05A-C5 |
| • RPS B TRIP UNIT IN CAL/FAIL       | P680-06A-D7 |
| • NS4 INBD ISOLATION OUT OF SERVICE | P601-18A-A3 |
| • BOP ISOL DW PRESS HIGH            | P601-19A-A6 |

There are no other surveillances in progress.

Which one of the following describes the required action to be performed if annunciator 1/2 SCRAM A/C occurs during performance of this surveillance?

- A. The alarm is 'unexpected'; notification of the Unit Supervisor is not required.
- B. The alarm is 'unexpected'; notification of the Unit Supervisor is required.
- C. The alarm is 'expected'; alarm confirmation is not required.
- D. The alarm is 'expected'; alarm confirmation is required.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 66

Which one of the following examples maintains the lowest total collective dose?

- A. One individual performing the job in a 60 mrem/hr field for 60 minutes.
- B. One individual installing temporary shielding in a 60 mrem/hr field for 30 minutes, performing the job in a 6 mrem/hr field for 60 minutes, and then removing the temporary shielding in a 6 mrem/hr field for 20 minutes.
- C. Two individuals performing the job in a 60 mrem/hr field for 35 minutes.
- D. Two individuals installing temporary shielding in a 60 mrem/hr field for 15 minutes, performing the job in a 6 mrem/hr field for 40 minutes, and then removing the temporary shielding in a 6 mrem/hr field for 10 minutes.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 67

Immediately following a reactor scram, the following plant conditions exist:

- Reactor power 0%
- RPV pressure 940 psig
- RPV water level +196 inches
- Two control rods are stuck Position 48
- Drywell pressure 0.5 psig
- Drywell temperature 130 °F
- Containment temperature 97 °F

Which of the following Plant Emergency Instruction(s) is/are required to be entered based on the plant conditions described above?

- A. PEI-T23, Containment Control and PEI-B13, RPV Control (Non-ATWS).
- B. PEI-T23, Containment Control and PEI-B13, RPV Control (ATWS).
- C. Only PEI-B13, RPV Control (ATWS).
- D. Only PEI-T23, Containment Control.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 68

The following plant conditions exist:

- A small-line break LOCA is in progress
- The reactor has been shutdown
- Drywell temperature is 250 °F and slowly increasing
- Drywell pressure is 2.0 psig and slowly increasing

Which one of the following PEI actions would not mitigate the increase in Drywell temperature and pressure, by itself?

- A. Restore NCC to Drywell coolers.
- B. Operate all available Drywell cooling.
- C. Perform a controlled depressurization of the RPV.
- D. Irrespective of cooldown rate, depressurize the RPV rapidly using the Main Turbine Bypass Valves.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 69

The plant is operating at 50% power.

Both Reactor Recirculation Pumps trip to OFF resulting in a core flow of 30 Mlbm/hr.

Which one of the following conditions will require the Control Room Operator to immediately insert a manual reactor scram in accordance with ONI-C51, Unplanned Change in Reactor Power or Reactivity?

- A.           APRM oscillations of 5% peak-to-peak are observed.
- B.           The pre-transient load line was 102%.
- C.           All OPRMs are currently inoperable.
- D.           As directed by Reactor Engineering.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 70**

The plant was operating at 35% power with the Reactor Recirculation Pumps in fast speed when a reactor scram occurred.

During the subsequent plant transient:

- RPV water level decreased to +152 inches before being restored to normal
- RPV pressure increased to 1074 psig before being restored to normal

Which one of the following describes the current Reactor Recirculation Pump breaker lineup?

CB 1A/B	CB 2A/B	CB 3A/B	CB 4A/B	CB 5A/B
------------	------------	------------	------------	------------

- |    |        |        |        |        |        |
|----|--------|--------|--------|--------|--------|
| A. | Closed | Closed | Closed | Closed | Open   |
| B. | Open   | Open   | Closed | Closed | Closed |
| C. | Closed | Closed | Open   | Open   | Open   |
| D. | Open   | Open   | Open   | Open   | Open   |

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 71

The following plant conditions exist:

- The Control Room has been evacuated due to toxic gas
- All actions of ONI-C61, Evacuation of the Control Room, have been completed
- Control has been transferred to the Division 1 Remote Shutdown Panel in accordance with IOI-11, Shutdown From Outside The Control Room

Assume no other operator actions have been performed.

Which set of plant systems will automatically initiate in response to a decreasing RPV water level condition?

- A. LPCS and LPCI 'A'
- B. LPCI 'A' and RCIC
- C. RCIC and HPCS
- D. HPCS and LPCS

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 72

The plant was operating at 100% power.

Combustible Gas Mixing Compressor 'A' was operating for its quarterly surveillance when the following simultaneous events occurred:

- The Balance-of-Plant (BOP) isolation valves isolated
- All standby ECCS Pumps started
- The Nuclear Closed Cooling System (NCC) isolated

Assuming RPV water level remained normal, which one of the following additional automatic actions immediately occurred?

- A. The RCIC System initiated.
- B. The Main Turbine tripped.
- C. The reactor scrammed.
- D. The MSIVs isolated.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 73

The plant is operating at 90% power.

Which one of the following describes the effect of an increase in air and non-condensable gases in the Main Condenser?

Overall plant efficiency will \_\_\_\_\_.

- A. decrease because condensate temperature decreases.
- B. decrease because the Main Turbine is operating with a higher backpressure.
- C. increase because condensate temperature increases.
- D. increase because the Main Turbine is operating with a lower backpressure.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 74

The plant is operating at 100% power.

A Control Room Operator inadvertently arms and depresses the LPCS & LPCI A MANUAL INITIATION pushbutton, 1E21A-S9, on panel H13-P601.

Which one of the following describes an effect of this inadvertent action?

- A. A loss of Drywell cooling occurs due to the loss of electrical power to the Drywell Cooling Fans.
- B. A loss of Drywell cooling occurs due to the isolation of cooling water flow to the Drywell Cooling Fans.
- C. A loss of Containment cooling occurs due to the loss of electrical power to the Containment Vessel Cooling Fans.
- D. A loss of Containment cooling occurs due to the isolation of cooling water flow to the Containment Vessel Cooling Fans.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 75

Which one of the following describes the LCO Bases for the Annulus Exhaust Gas Treatment System during MODES 1, 2, and 3 in accordance with Technical Specifications?

- A. Provides the capability to control the hydrogen concentration in the Drywell during a Design Bases Accident (DBA).
- B. Supports the operability of the Secondary Containment by maintaining the Secondary Containment at a positive pressure with respect to the environment.
- C. Processes radioactive material that leaks from the Primary Containment into the Secondary Containment prior to release during a Design Bases Accident (DBA).
- D. Limits the release of radioactive material to the environment during movement of recently irradiated fuel assemblies in the Primary Containment.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 76

A Loss of Coolant Accident has occurred.

The following plant conditions exist:

- Indicated Wide Range Level      +30 inches
- RPV pressure                        50 psig
- Containment temperature        260 °F
- Drywell temperature              275 °F

Which one of the following describes the status of the Wide Range level indication?

The Wide Range level indication is \_\_\_\_\_.

**PEI-SPI Figure 1a and 2a are provided for reference.**

- A.            not valid because indicated Wide Range level is less than 33 inches.
- B.            not valid because Drywell temperature is greater than RPV saturation temperature.
- C.            valid because indicated Wide range level is greater than 23 inches.
- D.            valid because Drywell temperature is less than RPV saturation temperature.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 77**

The plant is operating at 100% power.

The following plant conditions exist:

- SVI-E51-T2001, RCIC Pump and Valve Operability Test, is in progress
- The RCIC Turbine is in operation
- Suppression Pool temperature is 85 °F
- The Suppression Pool heatup rate is 12 °F per hour

Which one of the following is the maximum amount of time that the RCIC Turbine may continue to be operated without exceeding the Suppression Pool average temperature limit in accordance with Technical Specifications?

- A.            Approximately 30 minutes.
- B.            Approximately 50 minutes.
- C.            Approximately 100 minutes.
- D.            Approximately 120 minutes.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 78**

The plant is operating at 90% power.

The initial load line is 101%.

During a subsequent plant transient:

- HTR 6A EXST & INLET DRNS ISOL LEVEL HIGH alarm is received on panel H13-P870
- Feedwater temperature is slowly decreasing
- Reactor power is slowly increasing

Which one of the following describes the appropriate action which the Unit Supervisor should direct the Supervising Operator to perform?

- A. Maintain reactor power  $\leq 95\%$  using Cram Rods.
- B. Maintain reactor power  $\leq 95\%$  using Reactor Recirculation Flow Control.
- C. Lower reactor power to  $\leq 90\%$  using Cram Rods.
- D. Lower reactor power to  $\leq 90\%$  using Reactor Recirculation Flow Control.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 79

An off-site release event is in progress.

The following information is available for the Shift Supervisor:

- HIGH radiation alarm has been received on the TB/HB Ventilation GAS module, 1D17-K856
- TB/HB Ventilation GAS module indicates  $7.2 \times 10^5$  cpm
- TB/HB Ventilation GAS module HIGH alarm setpoint is  $3.4 \times 10^3$  cpm
- Chemistry reports that it will take 45 minutes to obtain a TB/HB Ventilation gas sample for analysis
- Chemistry reports that it will take 30 minutes to perform the Emergency Dose calculations needed to confirm the radiation levels at the site boundary

Which one of the following describes the required Emergency Plan classification for this event?

**EPI-A1 is provided for reference.**

- A. Unusual Event
- B. Alert
- C. Site Area Emergency
- D. General Emergency

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 80

Which one of the following is the minimum Reactor Cavity Pool water level required during movement of irradiated fuel assemblies within the RPV in accordance with Technical Specifications?

- A. 22 feet 9 inches above the top of the RPV flange.
- B. 22 feet 9 inches above the top of irradiated fuel assemblies seated within the RPV.
- C. 23 feet above the top of the RPV flange.
- D. 23 feet above the top of irradiated fuel assemblies seated within the RPV.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 81

The following plant conditions exist:

- The reactor scrammed on high reactor pressure
- MSIVs are isolated
- RPV water level 180 inches
- RPV pressure 975 psig
- Suppression Pool temperature 106 °F
- Suppression Pool level 18.0 ft

Which one of the following is the highest temperature the Suppression Pool can reach without exceeding the Heat Capacity Limit (HCL)?

**PEI-SPI Figure 4 is provided for reference.**

- A. 113 °F
- B. 115 °F
- C. 117 °F
- D. 119 °F



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 82**

During an ATWS, which one of the following identifies the highest Suppression Pool temperature, and its corresponding bases, that requires the initiation of the Standby Liquid Control System (SLC)?

- A. 110 °F; to ensure that Hot Shutdown Boron Weight is injected prior to Suppression Pool temperature exceeding the Heat Capacity Limit (HCL).
- B. 110 °F; to ensure that SLC is initiated prior to exceeding the Technical Specification limit that requires a reactor scram.
- C. 120 °F; to ensure that Hot Shutdown Boron Weight is injected prior to Suppression Pool temperature exceeding the Heat Capacity Limit (HCL).
- D. 120 °F; to ensure that SLC is initiated prior to exceeding the Technical Specification limit that requires RPV depressurization to less than 200 psig.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 83

PEI-B13, Emergency Depressurization, has been entered.

The following plant conditions exist:

- RPV water level -25 inches
- Suppression Pool level 5.0 ft
- Two (2) Non-ADS SRVs are open
- Main Condenser is not available

Which one of the following describes a method to depressurize the RPV?

**PEI-B13 Emergency Depressurization flowchart is provided for reference.**

- A. Open all ADS SRVs.
- B. Open six (6) ADS SRVs.
- C. Operate the RCIC System.
- D. Operate the Turbine Driven Feedwater Pumps (RFPTs).

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 84

Which one of the following conditions would require the Unit Supervisor to declare the Outboard Main Steam Isolation Valves (MSIVs) inoperable because they may be incapable of closing within their required Technical Specification times?

- A. Parallel Instrument Air header pressure less than 90 psig.
- B. ADS Air Storage Tank 'B' pressure less than 150 psig.
- C. ADS Air Storage Tank 'B' pressure less than 90 psig.
- D. Instrument Air header pressure less than 90 psig.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 85

The following plant conditions exist:

- The plant is in Day 4 of refueling outage RF08
- The RPV has been disassembled
- CORE ALTERATIONS have not commenced
- RPV water level is at the RPV flange
- The in-service RHR SDC Pump has just tripped
- Reactor coolant temperature is 120 °F

Assume no operator actions have been taken.

Which one of the following identifies the time available to restore shutdown cooling before RPV water level boils off to the Top of Active Fuel (TAF)?

**PDB-A0019 is provided for reference.**

- A.            Approximately 8.2 hours.
- B.            Approximately 12.4 hours.
- C.            Approximately 14.0 hours.
- D.            Approximately 21.0 hours.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 86**

The plant is operating at 100% power.

Replacement of an internal power supply will cause the complete de-energization of Division 1 of the Redundant Reactivity Control System (RRCS).

Which one of the following identifies the additional component/equipment that must be declared inoperable in accordance with Technical Specifications?

- A. SLC Subsystem 'A'.
- B. SLC Subsystem 'B'.
- C. RPS Trip System 'A'.
- D. RPS Trip System 'B'.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 87

The following plant conditions exist:

- CORE ALTERATIONS are in progress in reactor quadrant 'A'
- All control rods are fully inserted
- SRM Channel 'C' detector is fully withdrawn due to a failed detector drive

SRM Channel A/B Recorder, 1C51-R602A, has just failed and will have to be removed for repairs.

No other indications for SRM Channels 'A' and 'B' are currently available.

Which one of the following describes the Required Action, if any, that must be completed in accordance with Technical Specifications?

**Technical Specification 3.3.1.2, SRM Instrumentation is provided for reference.**

- A. Entry into LCO 3.3.1.2 is not required.
- B. Suspend CORE ALTERATIONS immediately.
- C. Restore the required SRM(s) to OPERABLE status within 4 hours.
- D. Place the Reactor Mode Switch in the SHUTDOWN position within 1 hour.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 88

Which one of the following conditions specifically requires SRO oversight of all RHR valve alignments and system manipulations when SHUTDOWN COOLING OTBD SUCT ISOL, 1E12-F008, and SHUTDOWN COOLING INBD SUCT ISOL, 1E12-F009, are open?

- A.       The RHR Pump Minimum Flow Valve, 1E12-F064A(B) is incapable of opening to provide minimum flow in MODE 3.
- B.       The RHR Pump Minimum Flow Valve, 1E12-F064A(B) is incapable of opening to provide minimum flow in MODE 4.
- C.       The Reference Leg Backfill System is out of service in MODE 3.
- D.       The Reference Leg Backfill System is out of service in MODE 4.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 89

The plant is operating at 100% power with both Reactor Feed Pump Turbines (RFPTs) on the Master Level Controller and the Motor Feed Pump (MFP) in Standby Readiness. The HPCS System is in Secured Status.

A Main Turbine trip with failure of the Bypass Valves occurs. All control rods do not fully insert.

The following panel H13-P680 indications exist:

- RRCS FDW RUNBACK annunciator is in alarm
- Amber RRCS FWRB RUNBACK INITIATED light is illuminated
- Amber MANUAL OVERRIDE AVAILABLE light is illuminated
- RPV water level is +100 inches and decreasing

As the Unit Supervisor, which one of the following operator actions should be directed in order to stabilize RPV water level using the Feedwater System?

- A. Operate a RFPT on its Manual Speed Control Dial.
- B. Operate a RFPT in MANUAL on the Startup Level Controller.
- C. Operate the MFP in MANUAL on the Startup Level Controller.
- D. Operate the MFP in MANUAL on its individual Flow Controller.



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 90

The plant is operating at 50% power when SB&PR Channel 'A' fails.

The SB&PR System automatically transfers to SB&PR Channel 'B'.

The plant remains operating at 50% power.

Which one of the following Power Distribution Limits Technical Specification LCOs is required to be entered?

- A. Linear Heat Generation Rate (LHGR).
- B. Minimum Critical Power Ratio (MCPR).
- C. Maximum Average Planar Ratio (MAPRAT).
- D. Average Planar Linear Heat Generation Rate (APLHGR).

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 91

Which one of the following conditions would allow the Unit Supervisor to direct the Safety System to be manually overridden?

- A. The LPCS System is required to maintain RPV water level above the top of active fuel.
- B. A valid HPCS initiation is confirmed by two independent RPV water level indications.
- C. The SPMU timer is running during a large break loss of coolant accident.
- D. A Diesel Generator is in operation without ESW cooling water flow.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 92

During reactor operation, one channel of the IRM System will be placed in an inoperable condition for the purpose of performing a channel functional test.

Provided that the associated IRM Function maintains RPS trip capability, which one of the following is the maximum time that entry into the associated Technical Specification Conditions and Required Actions may be delayed?

- A. 2 hours.
- B. 4 hours.
- C. 6 hours.
- D. 12 hours.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 93

The plant is in MODE 5 with the vessel head removed.

Which one of the following refueling activities shall be observed and directly supervised by a licensed SRO who has no other concurrent responsibilities?

- A. Re-seating of a fuel bundle during core verification.
- B. Movement of the Refueling Platform with all hoists unloaded.
- C. Normal withdrawal of a SRM detector for signal to noise checks.
- D. Normal withdrawal of a control rod when its associated core cell is de-fueled.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 94

Which one of the following describes how Technical Specification Completion Times are tracked when situations require entry into multiple Conditions within a single LCO?

- A. Only a single Completion Time is tracked for the Condition that was entered last.
- B. Only a single Completion Time is tracked for the Condition with the most limiting Completion Time.
- C. Separate Completion Times are tracked for each Condition starting from the time of discovery of each situation.
- D. Separate Completion Times are tracked for each Condition starting from the time of discovery of the initial situation.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 95**

The plant is in an outage. The Division 2 DG has been removed from service for maintenance. The Division 1 DG Room has been posted with a Shutdown Safety Sign.

During his daily plant tour, the Plant Manager wants to enter the Div 1 DG Room.

Which one of the following individuals must grant permission for the Plant Manager to enter the Division 1 DG Room in accordance with PAP-0116, Shutdown Safety?

- A. Unit Supervisor.
- B. Shift Supervisor.
- C. Shift Outage Director.
- D. Shutdown Safety Advisor.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 96

Which one of the following types of instruction changes requires the Unit Supervisor or Shift Supervisor to assign an expiration date in accordance with PAP-0522, Preparation, Review, and Approval of Procedures and Instructions?

- A. Intent Final Instruction Change.
- B. Non-Intent Final Instruction Change.
- C. Permanent Conditional Instruction Change.
- D. Non-Permanent Conditional Instruction Change.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 97

The following plant conditions exist:

- A Site Area Emergency has been declared
- You are the Shift Supervisor and Emergency Coordinator
- The TSC is still in the 'activation' process
- You have waived a plant worker's Federal 10CFR20 TEDE dose limit in order to perform a lifesaving activity in an emergency situation

Which one of the following is the recommended maximum emergency TEDE dose you can authorize the plant worker to receive in accordance with HPI-B0003, Processing of Personnel Dosimetry?

- A. 5 Rem
- B. 10 Rem
- C. 20 Rem
- D. 25 Rem



**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 98

The Plant Emergency Instructions (PEIs) have been entered.

As the Shift Supervisor, which one of the following sets of individuals are you required to notify in accordance with PAP-0201, Conduct of Operations?

**Note: A partial list will be incorrect.**

- A. System Central Dispatcher, NRC Resident Inspector, Plant Manager.
- B. Operations Manager, Shift Technical Advisor, NRC Resident Inspector.
- C. NRC Operations Center Duty Officer, Plant Manager, Operations Manager.
- D. Shift Technical Advisor, System Central Dispatcher, NRC Operations Center Duty Officer.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

**QUESTION 99**

The following plant conditions exist:

- MODE 3
- Unit 2 Startup Transformer is supplying Bus L10
- Division 2 DG is in Secured Status for pre-planned maintenance
- Unit 1 Startup Transformer needs to be removed from service due to a damaged bushing on the primary side of the transformer

Which one of the following electrical power sources will satisfy the requirement to have two OPERABLE qualified circuits between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System in accordance with LCO 3.8.1, AC Sources – Operating?

- A. Cross-tie Bus L10 with Bus L20.
- B. Restore the Division 2 DG to OPERABLE status.
- C. Feed Bus L10 from Bus L11 or L12 via the Unit 1 Auxiliary Transformer.
- D. Feed Bus L20 from Bus L21 or L22 via the Unit 2 Auxiliary Transformer.

**U.S. NUCLEAR REGULATORY COMMISSION  
WRITTEN EXAMINATION  
SENIOR REACTOR OPERATOR**

QUESTION 100

The following plant conditions exist:

- An earthquake has occurred
- SW PUMP DISCH HEADER PRESS LOW annunciator alarm on panel H13-P970 is in alarm
- ONI-P41, Loss of Service Water, has been entered.

As the Unit Supervisor, you have directed a Supervising Operator to start all three ESW loops as required by the Supplemental Actions of ONI-P41.

Which one of the following is the bases for this action?

- A. To maintain the ESW loops filled.
- B. To support the manual backwash of the ESW strainers.
- C. To prevent stagnant ESW forebay water from heating up.
- D. To maintain cooling water flow to safety-related components.