EXAMINATION ANSWER KEY

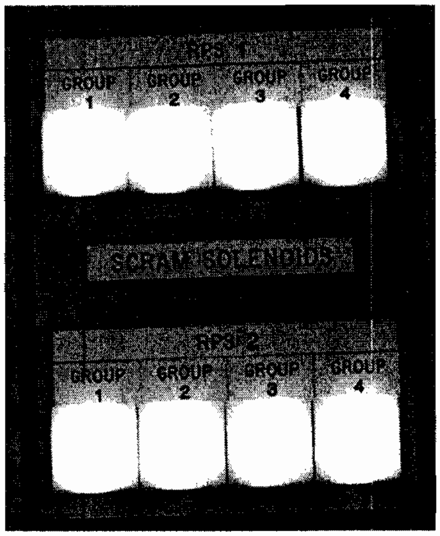
ILT 09-1 NRC RO Exam

1 **ID: 09-1 NRO1** **Points: 1.00**

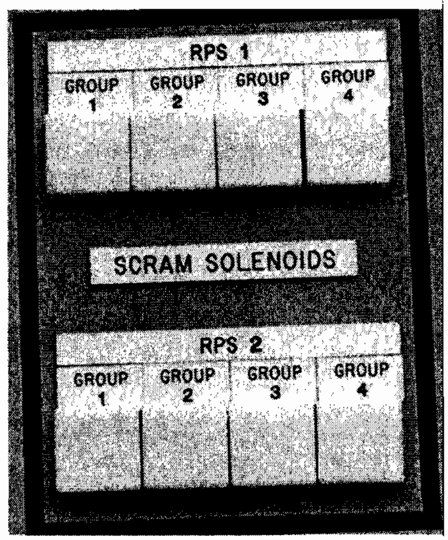
The plant was at rated power. Recirculation flow was at maximum.

Which of the following combination of **simultaneous** events will result in the indication below changing from **CONDITION 1** to **CONDITION 2**?

Condition 1



Condition 2



	Event 1	Event 2
A.	IRM 18 indicates upscale	APRM 8 indicates downscale
B.	APRM 4 indicates 114%	APRM 6 indicates 114%
C.	APRM 1 indicates INOP	APRM 6 indicates 120%
D.	APRM 3 indicates downscale	APRM 7 indicates INOP

Answer: C

Answer Explanation:

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

QID: 09-1 NRO1		
Question # / Answer	**1**	**Developer/Date: NTP 11/11/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
212000 RPS K1.01 Knowledge of the physical connections and/or cause- effect relationships between REACTOR PROTECTION SYSTEM and the following: Nuclear instrumentation					**3.7**	**3.9**
Level	**RO**	**Tier**	**2**	**Group**	**1**	

General References	**237E566**	**RAP-G1f RAP-G2f**	**RAP-G2f RAP-H7a**
Explanation	The plant is at rated power, and the question asks what events will cause all group scram solenoids to change state from currently energized to de-energized, as pictured. A 1/2 scram from RPS 1 APRMs (APRMs 1-4 HI-HI (>118%) or INOP) plus a 1/2 scram from RPS 2 APRMs (APRMs 5-8 HI-HI (>118%) or INOP) will cause all group scram solenoids to be de-energized. Correct answer C shows a 1/2 scram on RPS 1 from an inoperable RPS 1 APRM, plus a 1/2 scram on RPS 2 from an RPS APRM HI-HI. Answer A will result in a rodblock only, which does not impact the scram solenoids, and is incorrect. Answer B shows an RPS 1 and RPS 2 APRM above the rodblock setpoint of 113%, but below the scram setpoint of the HI-HI. Again, RPS in not impacted, and answer B is incorrect. Answer D shows an RPS 1 APRM downscale, which only provides a rod block, plus an RPS 2 APRM INOP resulting in a 1/2 scram on RPS 2 only. Thus, only RPS 2 group scram solenoids are affected but not all group scram solenoids are affected. Answer D is incorrect.		

References to be provided during exam:	**None**	

Learning Objective	2621.828.0.0029 LO 215-10446

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

<table>
<tr><td colspan="3">Question Source (New, Modified, Bank)</td><td colspan="2">New</td></tr>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X
3:SPK</td></tr>
<tr><td colspan="4">NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td>5</td><td>55.43</td><td></td></tr>
<tr><td colspan="4">(SRO Only)</td></tr>
<tr><td colspan="5">Time to Complete: 1-2 minutes</td></tr>
</table>

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

2 **ID: 09-1 NRO2** **Points: 1.00**

The plant was at rated power when an event occurred. The Panel Operator reports the following observations:

- ISOL COND - LOGIC TRAIN ACTUATED I annunciator is in alarm
- ISOL COND - LOGIC TRAIN ACTUATED II annunciator is in alarm
- EDG 1 UNIT IDLING light is energized
- EDG 1 UNIT START light is de-energized
- EDG 2 UNIT IDLING light is energized
- EDG 2 UNIT START light is de-energized
- RPV pressure indicates 425 psig and lowering slowly

Which of the following is correct?

A. **All** 4 RWCU Isolation Valves indicate **closed**.

B. **All** 4 Core Spray Parallel Isolation Valves indicate **open**.

C. **Both** EDG 1 and EDG 2 Output Breakers indicate **closed**.

D. **All** 10 RB HVAC MAIN SUPPLY HEADER VALVES indicate **open**.

Answer: A

Answer Explanation:

QID: 09-1 NRO2		
Question # / Answer	**2**	**Developer/Date: NTP 11/11/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
223002 PCIS/Nuclear Steam Supply Shutoff K1.02 - Knowledge of the physical connections and/or cause- effect relationships between PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF and the Reactor water cleanup				**3.3**	**3.5**
Level	**RO**	**Tier**	**2**	**Group**	**1**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

General References	EMG-SP1	341 RAP-C1a	330 2621.828.0.0013
Explanation	The plant was at rated power when an event occurred which resulted in the following (from the indications provided): both Isolation Condensers are in service and both EDGs have idle started. The ICs will auto initiate from either an RPV water level lo-lo (90") or RPV high pressure (1051 psig). The EDGs will auto idle start from either an RPV water level lo-lo (86"), a high Drywell pressure (3 psig) or from Low Lube Oil Temperature. The only single event that will both initiate the ICs and idle start the EDGs is an RPV water level lo-lo. On an RPV water level lo-lo, all 4 RWCU isolation valves will close if open (not all 4 valves close on an RWCU system upset condition). Answer A is correct. Under the conditions provided, the Core Spray System has auto initiated. But since RPV pressure is 425 psig, the Core Spray Parallel Isolation Valves are still closed and will auto open when RPV pressure lowers to 305 psig. Answer B is incorrect. The conditions provided show only that the EDGs have idle started and are not supplying their respective emergency busses. Therefore, the EDG output breakers are open. Answer C is incorrect. The conditions provided show that the Standby Gas treatment System has auto started and normal RB HVAC has secured. Answer D is incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.828.0.0039 LO 204-10445		

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:PEO
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

3 **ID: 09-1 NRO3** **Points: 1.00**

Which of the following correctly states the **normal** power supply to USS 1E1?

A. R144 Line

B. Transformer SA

C. Transformer SB

D. North Yard Distribution J69361

Answer: A

Answer Explanation:

QID: 09-1 NRO1		
Question # / Answer	**1**	**Developer/Date: NTP**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
262001 AC Electrical Distribution K2.01 - Knowledge of electrical power supplies to the following: Off-site sources of power					**3.3**	**3.6**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**BR 3000**		**BR 3002 sh. 4**			
Explanation	The R144 line is the normal offsite power supply to USS 1E1, and the J69361 Line is the alternate offsite power supply. Answer A is correct. All other answers are off-site power supplies, just not to USS 1E1.					
References to be provided during exam:	**None**					
Learning Objective	2621.828.0.016B LO 262-10435					

Question Source (New, Modified, Bank)	New

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Cognitive Level	Memory or Fundamental Knowledge	X 1:F	Comprehension or Analysis	
	NUREG 1021 Appendix B: Fact			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

4 **ID: 09-1 NRO4** **Points: 1.00**

The plant was at power when the following annunciator alarmed:

- DC PWR LOST - BUS A/B UV

The Operator reports 0 Volts on 125 VDC Bus A.

Which of the following is correct under the given condition?

A. The amplidyne **cannot** regulate the generator output voltage.

B. Feedwater Pump A **cannot** be tripped from the Control Room.

C. The turbine **cannot** be manually tripped from the Control Room.

D. The annunciators to Control Room Panels 1F/2F and 3F **cannot** function.

Answer: A

Answer Explanation:

QID: 09-1 NRO4		
Question # / Answer	4	Developer/Date: NTP 11/12/09

Knowledge and Ability Reference Information							
K&A						Importance Rating RO	Importance Rating SRO
263000 DC Electrical Distribution K2.01 - Knowledge of electrical power supplies to the following: Major D.C. loads						3.1	3.4
Level	RO	Tier	2	Group	1		
General References	ABN-53		BR 3028				

Explanation	The plant was at rated power when indications show a loss of volts to 125 VDC Bus A. When this occurs, generator voltage control transfers to the manual rheostat and the amplidyne can no longer control generator output voltage. Answer A is correct. Control power for Feedwater Pump A breaker is supplied by 125 VDC Bus C and is not impacted by the event. Answer B is incorrect. 125 VDC Bus A provides the normal power to 125 VDC Power Panel E. If voltage were lost to this panel, the ability to trip the turbine is impacted. But, when a loss of voltage is sensed from DC-A, the alternate DC supply from 125 VDC Bus B will automatically pickup Power Panel E, and Power Panel E remains energized. Answer C is incorrect. Also, if Power Panel E were to become de-energized, control room annunciators will not function. But since the alternate DC supply will automatically pickup Power Panel E, it remaine energized and control room annunciation is not impacted. Answer D is incorrect.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0012 LO 1106

Question Source (New, Modified, Bank)				**Modified**
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:F**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Fact			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

5 **ID: 09-1 NRO5** **Points: 1.00**

The plant was at rated power when the applied voltage to LPRM 20-49D was lost. (LPRM 20-49D inputs into APRM 6)

Which of the following states the impact on APRM 6 indicated reactor power and on reactor power indication provided by heat balance?

	APRM 6 Power Indication	Heat Balance Power Indication
A.	Indicates lower	Indicates lower
B.	Indicates lower	No impact
C.	No impact	Indicates lower
D.	No impact	No impact

Answer: B

Answer Explanation:

QID: 09-1 NRO5		
Question # / Answer	5	Developer/Date: NTP 11/12/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
215005 APRM/LPRM K3.08 - Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: Core thermal calculations					3.0	3.4
Level	RO	Tier	2	Group	1	
General References	GFES		NF-AB-770		LP 2621.828.0.0029	

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	When the applied voltage is lost to the LPRM detector, it can no longer collect all the generated ion pairs and the LPRM output will go down. As this single LPRM output lowers, APRM 6 indication will also lower since the LPRM is in its normal state and not bypassed from the APRM. The heat balance on the other hand, is not affected by the number of neutron counts and will remain the same since there is no change in reactor power. Answer B is correct and answer A is incorrect. The other answers are plausible if the candidate does not understand neutron detector operation, how the APRM is affected by LPRM inputs or heat balance calculations. The APRM would show no impact if the LPRM were bypassed. Answers C & D are incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0029 215-10453	

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

6 **ID: 09-1 NRO6** **Points: 1.00**

The plant is at 5% power during a startup. The steam flow signal from the Digital Feedwater Control System suddenly drifts up to 50% steam flow.

Which of the following states the impact on the Rod Worth Minimizer (RWM)?

A. The RWM will no longer enforce compliance with the control rod sequence.

B. The RWM will insert both an insert block and a withdraw block due to the RWM Fault.

C. The RWM will continue to enforce compliance with the control rod sequence until the Low Power Setpoint is reached.

D. The RWM will automatically shift to the Power Operations Mode to enforce compliance with the control rod sequence.

Answer: A

Answer Explanation:

QID: 09-1 NRO6		
Question # / Answer	6	Developer/Date: NTP 11/12/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
259002 Reactor Water Level Control K3.03 - Knowledge of the effect that a loss or malfunction of the REACTOR WATER LEVEL CONTROL SYSTEM will have on following: Rod worth Minimizer					2.7	2.9
Level	RO	Tier	2	Group	1	
General References	409		2621.828.0.0041			

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant is starting up at 5% power. The RWM is in service in the low power mode, enforcing the control rod sequence. The RWM determines reactor power level by the amount of steam flow, which comes from the Digital Feedwater Control System. In the current low power mode, the RWM will enforce the control rod sequence up to the low power setpoint (LPSP), which is set at 35% power. When the steam flow to the RWM rises to 50%, the RWM recognizes that reactor power is above the LPSP and that it will no longer enforce compliance to the control rod sequence. Answer A is correct. The RWM does have a high power operations mode, which when activated manually, will also enforce compliance to the control rod sequence. There is no automatic swap between the RWM modes. A failure of control rod position information can result in both an insert and withdraw block. The other answers are plausible but incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0041 LO 217-10444	

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

7 **ID: 09-1 NRO7** **Points: 1.00**

The plant was at rated power when an MSIV closure event occurred. RPV pressure **peaked** at 1215 psig.

Which of the following is correct regarding the operation of the EMRVs and the RPV Safety Relief Valves (SRV) at the peak pressure?

	EMRV Operation	SRV Operation
A.	**All** EMRVs opened through actuation of a pressure switch	5 SRVs opened through actuation of a pressure switch
B.	**All** EMRVs opened from RPV pressure overcoming the valve spring pressure	4 SRVs opened through actuation of a pressure switch
C.	**All** EMRVs opened from RPV pressure overcoming the valve spring pressure	5 SRVs opened from RPV pressure overcoming the valve spring pressure
D.	**All** EMRVs opened through actuation of a pressure switch	4 SRVs opened from RPV pressure overcoming the valve spring pressure

Answer: D

Answer Explanation:

QID: 09-1 NRO7		
Question # / Answer	7	**Developer/Date: NTP 11/12/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

239002 SRVs **K4.08 - Knowledge of RELIEF/SAFETY VALVES design feature(s) and/or interlocks which provide for the following: Opening of the SRV from either an electrical or mechanical signal**	**3.6**	**3.7**

Level	**RO**	**Tier**	**2**	**Group**	**1**

General References	**420**	**UFSAR 5.2.2**	**729E182**
Explanation	The plant was at rated power when the MSIVs closed and RPV pressure peaked at 1215 psig. 2 EMRVs open at 1085 psig (TS value) and 3 open at 1105 psig. 4 SRVs open at 1212 psig and 5 open at 1221 psig. Therefore at an RPV pressure of 1215 psig, all EMRVs and 4 SRVs have opened. The EMRVs open for ADS (Automatic Depressurization System) and in the Pressure Relief Mode. In the pressure relief mode, the EMRVs open to protect the RPV from an over-pressure condition. But, regardless if EMRV is opened manually, for the ADS function, or in the pressure relief mode, the EMRV solenoid must be energized to open the valve. In the pressure relief mode, a pressure switch will actuate to energize the solenoid. If there were no power available to the EMRV solenoids, the EMRVs will not open, regardless of the RPV pressure. The SRVs on the other hand, are purely mechanical. When RPV pressure overcomes the SRV spring pressure, the SRV will open, and can open with no electrical power required. Therefore, all EMRVs have open from actuation of the pressure switch, and 4 SRVs have opened since RPV pressure has overcome the valve spring pressure. Answer D is correct. The other answers are plausible if the candidate does not understand the operating mechanism of the valves or confuses the valve open setpoints. All other answers are incorrect.		

References to be provided during exam:	**None**	

Learning Objective	2621.828.0.0005 LO 368

Question Source (New, Modified, Bank)	**New**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

<table>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X
3:SPK</td></tr>
<tr><td colspan="4">NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td>7</td><td>55.43</td><td></td></tr>
<tr><td colspan="4">(SRO Only)</td></tr>
<tr><td colspan="5">Time to Complete: 1-2 minutes</td></tr>
</table>

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

8 **ID: 09-1 NRO8** **Points: 1.00**

The plant is at rated power.

Which of the following states an automatic pump start from the associated listed parameter? (Assume none of the listed pumps are currently running)

	Auto Pump Start	**Parameter**
A.	RBCCW Pump 1	Low RBCCW System flow
B.	TBCCW Pump 2	Low TBCCW System pressure
C.	Service Water Pump 1-1	Breaker position of Service Water Pump 1-2
D.	CRD Pump B	Loss of volts to 4160V Bus 1C

Answer: B

Answer Explanation:

QID: 09-1 NRO8		
Question # / Answer	**8**	**Developer/Date: NTP 11/12/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
400000 Component Cooling Water K4.01 - Knowledge of CCWS design feature(s) and or interlocks which provide for the following: Automatic start of standby pump				**3.4**	**3.9**
Level	**RO**	**Tier**	**1**	**Group**	**2**
General References	**RAP-Q1f**				

Explanation	The plant is in a normal configuration at rated power. When the TBCCW System pressure drops to 79 psig and after a small time delay (10 seconds), any non-running TBCCW is signaled to start. Answer B is correct. All other pumps will start if voltage is lost to its respective 4160V emergency bus and loading by its respective EDG. RBCCW Pump 1 has no auto start on low system flow, but does if voltage is lost to USS 1A2 (fed from Bus 1C and EDG 1). When voltage is lost, there will be no RBCCW flow and EDG 1 will pickup USS 1A2 and the pump resumes flow. Answer A is incorrect but plausible. SW Pump 1-1 is independent on the breaker position of the other SW pump. SW Pump starts similar to RBCCW. Answer c is incorrect but plausible. CRD Pump B will auto start from a loss of volts to 4160V Bus 1D, not Bus 1C. Answer D is incorrect but plausible.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0048 LO 274-10443	

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:F**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Fact			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

9 **ID: 09-1 NRO9** **Points: 1.00**

The plant is shutdown and is cooling down using the Shutdown Cooling System (SDC). Present plant conditions include the following:

- Shutdown Cooling Loops A and B are in service
- RECIRC PUMP SUCTION TEMPS indicates 175 °F and lowering
- A, B, C and D Reactor Recirculation Loops are in an Idle configuration
- Recirculation Pump E is operating

Which of the following will result in the **greatest** impact on the RPV cooldown rate?

A. SDC Loop A DISCHARGE V-17-55 indicates 0%.

B. Shutdown Cooling Loop B senses 3 psig suction pressure.

C. Reactor Recirculation Pump E trips and the loop is placed in an Idle configuration.

D. Shutdown Cooling System Inlet Isolation Valve V-17-19 indicates red and green lights on.

Answer: D

Answer Explanation:

QID: 09-1 NRO9		
Question # / Answer	9	**Developer/Date: NTP 11/14/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
205000 Shutdown Cooling K5.02 - Knowledge of the operational implications of the following concepts as they apply to SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) : Valve operation					2.8	2.9
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**BR E1129 GE 148F711**		**GE 157B6350 sh. 157A**		**RAP-C2d, -C3d 305**	

Explanation	The plant is shutdown and is cooling down with 2 SDC loops. If the SDC inlet isolation valve V-17-19 comes off the full open position, all SDC pumps are interlocked to trip. Thus, with the red and green lights on, SDC pumps will trip resulting in a total loss of SDC and there will be no cooling of the RPV coolant and the cooldown rate will cease. Answer D is correct. When the SDC loop A discharge valve indicates 0%, the valve is closed. This SDC loop is no longer providing any cooling. But with the SDC loop B still in service, there is still about 50% of the initial amount of SDC cooling the RPV. Answer A is incorrect. When SDC loop B senses 3 psig suction pressure, the SDC B pump will trip. This results in 50% of the initial amount of SDC cooling the RPV. Answer B is incorrect. The SDC System takes a suction on Recirculation Pump E suction piping and discharges to the discharge piping of Recirculation Pump E. When the recirculation Pump trips, SDC could short cycle through the tripped pump instead of through the RPV, and would result in less cooling and a reduced RPV cooldown. But with the recirculation pump in an idle condition, its discharge valve is closed, which is the an allowable configuration and short cycling is not a concern. Answer C is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0045 LO 205-10445	

Question Source (New, Modified, Bank)				**Modified**
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:DR**
	NUREG 1021 Appendix B: Describe or recognize relationships			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

10 **ID: 09-1 NRO10** **Points: 1.00**

The plant is at rated power. The Operator is starting EDG 1 for Peaking Operation IAW procedure 341, Emergency Diesel Generator Operation, at the EDG Switchgear.

IAW procedure 341, which of the following actions and/or observations are required to parallel the EDG once it has been started?

	EDG 1 Output Voltage	EDG 1 Output Frequency	Synchroscope Position When EDG Output Breaker is Manually Closed by the Operator
A.	Slightly **higher** than line voltage	Slightly **higher** than line frequency	11 o'clock position
B.	Slightly **lower** than line voltage	Slightly **lower** than line frequency	12 o'clock position
C.	Slightly **higher** than line voltage	Slightly **lower** than line frequency	11 o'clock position
D.	Slightly **lower** than line voltage	Slightly **higher** than line frequency	12 o'clock position

Answer: A

Answer Explanation:

QID: 09-1 NRO10		
Question # / Answer	**10**	**Developer/Date: NTP 11/14/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

264000 EDGs K5.05 - Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) : Paralleling A.C. power sources					**3.4**	**3.4**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	341					
Explanation	The plant is at rated power with an operator starting EDG 1 at the local switchgear IAW procedure 341. To synchronize the EDG to the line, EDG output voltage must be slightly higher than line voltage, the synchroscope must be moving slowly in the fast direction. This means that the EDG output frequency is slightly higher than line frequency. The EDG output breaker will then be closed when the synchroscope hand reaches the 11 o'clock position. Answer A is correct. All other answers are plausible if the candidate does not know the relationships of voltage/frequency while paralleling two generator sources, but are incorrect in some fashion.					
References to be provided during exam:	**None**					
Learning Objective	2621.828.0.0013 LO 264-10446					

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

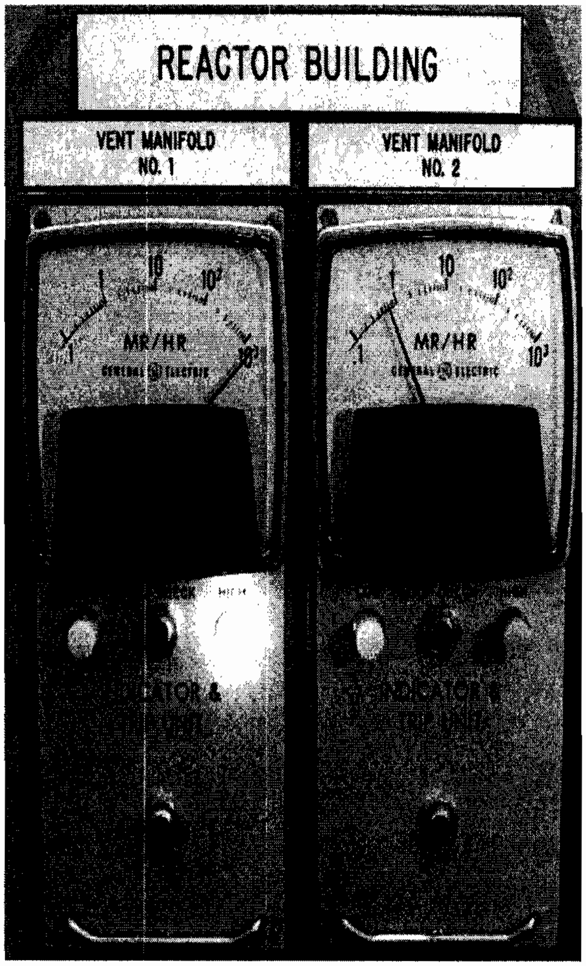
11 **ID: 09-1 NRO11** **Points: 1.00**

The plant is at rated power. The STANDBY GAS SELECT switch is in position SYS 2.

An event occurred which resulted in the following annunciator alarming:

- RADIATION MONITORS PROCESS RX BLDG - VENT HI

The Operator observed the following indications:



Which of the following shows the **first** response of the Standby Gas Treatment System (SGTS), if any?

A.


OFF
EXHAUST FAN
HAND
OFF
AUTO
STANDBY GAS 1
INLET
V-28-23
ORIFICE INLET
V-28-24
OUTLET
V-28-26
INLET AND OUTLET
VALVES
OFF
EXHAUST FAN
HAND
OFF
AUTO
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OUTLET
INLET AND OUTLET
VALVES

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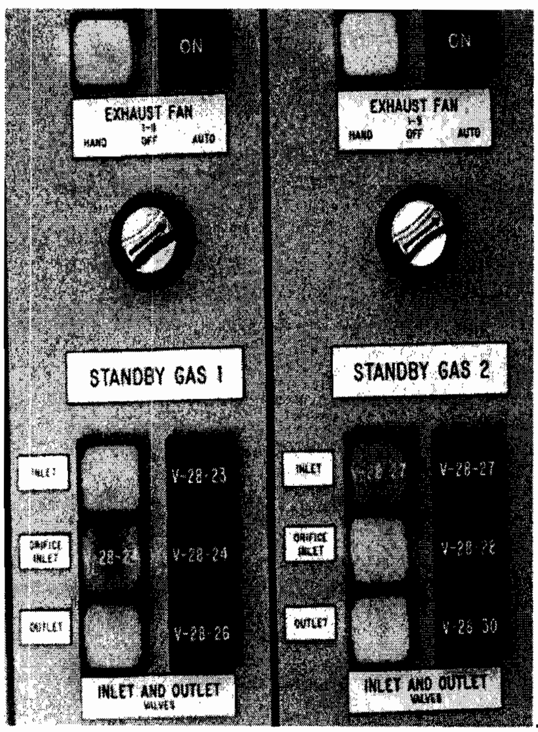
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VALVES

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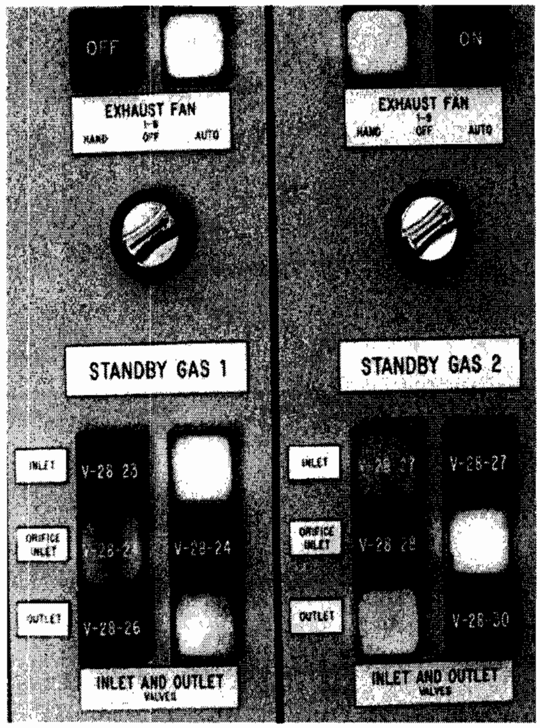
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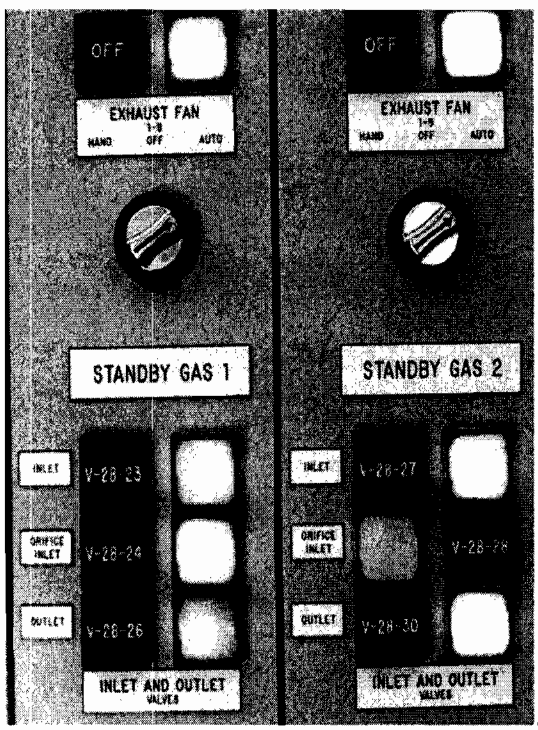
V-28-30
INLET AND OUTLET
VALVES

D.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OUTLET
V-28-23
V-28-24
V-28-26
V-28-27
V-28-30
INLET AND OUTLET
VALVES

Answer: A

Answer Explanation:

QID: 09-1 NRO11		
Question # / Answer	11	Developer/Date: NTP 11/14/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
261000 SGTS K6.04 - Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM : Process radiation monitoring					2.9	3.1
Level	RO	Tier	2	Group	1	
General References	330		RAP-10F1f			
Explanation	The plant is at rated power when an event occurs. The provided indications shows that RB Vent Rad Monitor #1 has failed upscale. The logic to auto start SGTS is 1 rad monitor above the setpoint (9 mr/hr). Therefore, the SGTS has received an initiation signal. When the auto start signal is received, both SGTS fans start and SGTS valves for both fans re-align, as shown in Answer A. Answer A is correct. Since SGTS 1 is the selected system, then after flow has been established in system 1, SGTS System 2 will shutdown and re-align as shown in Answer C. This re-alignment occurs in about 2-3 minutes after the start signal is received. Since the question asks for the correct indications after only 1 minute, then the system alignment still looks as in Answer A. Answer C is incorrect. Answer B shows SGTS in its standby condition. If the candidate does not understand the logic to start SGTS, this could be selected. Answer B is incorrect. Answer D shows a start of both SGTS fans but System has a broken shaft but remains in service and the orifice valve for System 2 has closed since it is the only fan drawing a suction. Answer D is incorrect.					
References to be provided during exam:	None					

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Learning Objective	2621.828.0.0042 LO 261-10445

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event of outcome.			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

12 **ID: 09-1 NRO12** **Points: 1.00**

The plant was at rated power when the following annunciator alarmed:

- 1B2 MN BREAKER TRIP

Which of the following states the impact on the Vital AC System?

A. CIP-3 auto transfers to USS 1A2.

B. VACP-1 auto transfers to VMCC 1A2.

C. PAIPP 2 must be manually transferred to VMCC 1A2.

D. MCC 1AB2 must be manually transferred to VMCC 1A2.

Answer: B

Answer Explanation:

QID: 09-1 NRO12		
Question # / Answer	12	Developer/Date: NTP 11/16/09

Knowledge and Ability Reference Information						
K&A					Importance Rating RO	Importance Rating SRO
262002 UPS (AC/DC) K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : AC electrical power					2.7	2.9
Level	RO	Tier	2	Group	1	
General References	ABN-48		RAP-9XF3c		3013 sh. 1 3002 sh. 2	

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant was at rated power when indications of the loss of USS 1B2 appeared. USS 1B2 powers VMCC 1B2. VMCC normally powers MCC 1AB2, the rotary inverter which supplies CIP-3, and VACP-1. When power is lost to VACP-1, an auto transfer switch operates to power VACP-1 from VMCC 1A2. Answer B is correct. The inverter to CIP-3 will transfer to a DC power supply (and VMCC as a backup). Answer A is incorrect. USS 1B2 power PAIPP 2, and it power supply is not transferable to another supply. Answer c is incorrect. When power is lost to MCC 1AB2, an auto transfer switch operates to power MCC 1AB2 from VMCC 1A2. Answer D is correct.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0056 LO 262-10445	

Question Source (New, Modified, Bank)				**Bank**
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

13 **ID: 09-1 NRO13** **Points: 1.00**

The plant was starting up after an outage. RECIRC PUMP SUCTION TEMPS indicates 350 °F.

The Operator notes a rising Drywell pressure and reports Drywell Unidentified leakage of 50 GPM and steady.

5 minutes later, the following annunciators alarmed **simultaneously**:

- LKOUT RELAY 86/S1A TRIP
- LKOUT RELAY 86/S1B TRIP
- DW PRESS HI-HI RV 46 A/B
- DW PRESS HI-HI RV 46 C/D

Which of the following states the EDG 1 loading sequence and the impact on RPV water level 60 seconds **after** EDG 1 output breaker closes? (**Note**: the pumps are listed in order of starting sequence)

	EDG 1 Loading Sequence	Impact on RPV Water Level
A.	Core Spray Main Pump C Core Spray Booster Pump C CRD Pump NC08B	Level is rising
B.	Core Spray Main Pump A Core Spray Booster Pump A CRD Pump NC08A Service Water Pump 1-1	Level is steady
C.	Core Spray Main Pump A Core Spray Booster Pump A CRD Pump NC08A	Level is rising
D.	CRD Pump NC08B Core Spray Main Pump C Core Spray Booster Pump C	Level is steady

Answer: C

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer Explanation:

QID: 09-1 NRO13		
Question # / Answer	13	Developer/Date: NTP 11/16/09

<table>
<tr><th colspan="6">Knowledge and Ability Reference Information</th><th></th></tr>
<tr><td colspan="5" rowspan="2">K&A</td><td colspan="2">Importance Rating</td></tr>
<tr><td>RO</td><td>SRO</td></tr>
<tr><td colspan="5">209001 LPCS
A1.07 - Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Emergency generator loading</td><td>3.0</td><td>3.1</td></tr>
<tr><td>Level</td><td>RO</td><td>Tier</td><td>2</td><td>Group</td><td colspan="2">1</td></tr>
<tr><td>General References</td><td colspan="2">341</td><td colspan="2">201 Att. 7</td><td colspan="2">RAP-S1b
RAP-C1f
RAP-B2e</td></tr>
<tr><td>Explanation</td><td colspan="6">The plant was starting up with coolant temperature at 350 °F (120 psig). Drywell pressure began to rise from a leak of 50 gpm (steady). 5 minutes later, Drywell pressure reached the LOCA signal setpoint, simultaneous with a loss of offsite power.
Both EGDs start and close onto their respective emergency bus (in < 10 seconds). With a LOOP/LOCA, EDG1 loads as follows: lighting/instrument bus, Core spray main Pump NZ01A, Core Spray Booster Pump NZ03A, and CRD Pump A. For a non-LOOP LOCA, Service Water and RBCCW also start.
Since RPV pressure is below the opening setpoint for the Core Spray Parallel Isolation Valves (305 psig), these valves open. With 2 sets of Core Spray injecting into the RPV (plus the already running Feed/Condensate System), RPV water level will rise. Answer C is correct.
Answer A lists loads on EDG 2 and is incorrect.
Answer B states a component on EDG 1 which would auto start from just a LOOP (and no LOCA). It also has the incorrect RPV water level impact. Answer B is incorrect.
Answer D has loads which are powered from EDG 1, but sequence is incorrect, as the water level impact. Answer D is incorrect.</td></tr>
</table>

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

References to be provided during exam:	**Attachment 201-7**	
Learning Objective	2621.828.0.0013 LO 264-10444	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**x 3:SPR**
	NUREG 1021 Appendix B: Solve a problem using a reference			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

14 **ID: 09-1 NRO14** **Points: 1.00**

The plant was at rated power when the following annunciator alarmed:

- SV/EMRV NOT CLOSED

The Operator reported the following indications:

- EMRV NR108A **and** NR108B indicate in the VALVE OPEN REGION
- **All** AUTO DEPRESS VALVE green lights are lit
- EMRV DISCH NR108A, B & E (Panel 1F/2F) indicates 328 °F
- MWe has dropped by 30 MWe

Which of the following states the status of the EMRVs?

A. **Both** EMRVs are full open, as evidenced by the magnitude of the drop in electrical output.

B. **Either** or **both** EMRVs are open, but must be confirmed at the EMRV Tailpipe Temperature Indicator (RB 23').

C. **Both** EMRVs are open as evidenced by the VALVE OPEN REGION and Panel 1F/2F EMRV DISCH temperature.

D. **Either** or **both** EMRVs are open, but must be confirmed by the red HI-ALARM lights in the Acoustic Monitoring Panel (15R).

Answer: B

Answer Explanation:

QID: 09-1 NRO14		
Question # / Answer	14	**Developer/Date: NTP 11/16/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

218000 ADS **A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATIC DEPRESSURIZATION SYSTEM controls including: ADS valve tail pipe temperatures**						**3.4**	**3.6**
Level	**RO**		**Tier**	**2**	**Group**	**1**	
General References	**RAP-B4g**		**ABN-40**			**UFSAR Table 5.1-1**	
Explanation	The plant is at power when indications show that one or both EMRVs have opened (by other than solenoid energization). When a single EMRV opens, the VALVE OPEN REGION will be indicated for that EMRV and for another EMRV located physically close-by. EMRVs A, B & E have a common monitored discharge tailpipe, which will show an elevated temperature for any of the 3 EMRVs being open. Thus, from the Panel 1F/2F tailpipe temperature indications, it will not show conclusively which valve is open. Individual EMRV tailpipe indicators on RB23' will show elevated temperatures for the individually open valves. Thus one or both EMRVs are open, but must be determined from the individual tailpipe indications locally. Answer B is correct. The number of open EMRVs cannot be ascertained from the drop in electrical power. A single EMRV full open, passes about 600,000 lb/hr of steam at 1250 psig, which represents about 8% power, or about 50 MWe. If both valves were open 50%, this would provide about the same load drop for a single valve open 100%. Thus load drop alone cannot be used to tell how many EMRVs are open. Answer A is incorrect. As stated, EMRVs A & B have a common tailpipe discharge with a common temperature indication and cannot be used to tell which is open. Also, as stated it is common for 1 open EMRV to influence the acoustic monitor of a near-by EMRV such that both indicate in the open region. Answer C is incorrect. Under the conditions provided, the Hi-ALARM lights for EMRVs A & B will be lit, but cannot be used to determine which valves are open. Answer D is incorrect.						
References to be provided during exam:		**None**					

Learning Objective	2621.828.0.0005 LO 7319

Question Source (New, Modified, Bank)				New
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describing or recognizing relationships			
10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

15 **ID: 09-1 NRO15** **Points: 1.00**

The plant was starting up after an outage. Present plant conditions include the following:

- REACTOR MODE SELECTOR switch is in STARTUP
- **No** ½ scrams or rodblocks currently exist
- **All** IRMs Recorders are on the 0-125% scale in Range 2
- Control rod withdrawals have been on-hold for the last 20 minutes

Several annunciators then alarmed. The Operator reported that **all** RPS Group Solenoids indicate energized.

The IRM indications 20 minutes ago, **and** the IRM indications now **after** the annunciators were received are shown below:

20 minutes ago:

IRM	Individual IRM Indicating Lights LIT (Panel 4F)	IRM Recorder Reading
11	None	8
12	None	9
13	HI-HI HIGH DN SCL OR INOP	90
14	None	10

After the annunciators were received:

IRM	Individual IRM Indicating Lights LIT (Panel 4F)	IRM Recorder Reading
11	DN SCL OR INOP	6
12	None	8
13	HI-HI HIGH DN SCL OR INOP	125
14	None	9

Which of the following states the plant response from this event and the required Operator action?

	Plant Response	Operator Action
A.	Rodblock from IRM 11 INOP	Bypass IRM 11
B.	Rodblock and 1/2 scram from IRM 13 INOP	Bypass IRM 13
C.	Rodblock and 1/2 scram from IRM 13 upscale	Range up on IRM 13
D.	Rodblock from IRM 11 downscale	Range down on IRM 11

Answer: D

Answer Explanation:

QID: 09-1 NRO15		
Question # / Answer	**15**	**Developer/Date: NTP 11/17/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
215003 IRM A2.04 - Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Up scale or down scale trips						**3.7**	**3.8**
Level	**RO**	**Tier**	**2**	**Group**	**1**		
General References	**RAP-G4e**	**RAP-H7a**	**RAP-H7a**				

Explanation	The plant is starting up after an outage and IRM 11-14 indications are provided both from 20 minutes ago and after the annunciator was received. A comparison of the 2 sets of indications shows that IRM11, 12 & 14 each went down slightly. IRM 11 picked up the downscale or inop light. IRM 14 went up to fill scale with no change in indicating lights. IRM 11 has the DN SCL or INOP light lit. A 7% on the 0-125% scale would account for this light being lit (setpoint for downscale is <7% on the 0-125% scale). No other indications are provided which show that IRM 11 is INOP. Also, an INOP IRM would produce a 1/2 scram and the RPS 1 group solenoids would be de-energized. Therefore, IRM 11 is reading below the downscale setpoint, and this would impose a control rod block (except when the IRM is on Range 1). Thus, IRM 11 is downscale causing a rodblock. A downscale IRM will cause the DN SCL OR INOP light to be lit. To correct this, the Operator must range down on the IRM. Answer D is correct. An INOP IRM 11 would cause a 1/2 scram which de-energizes the RPS 1 scram solenoids. Since RPS is normal, no 1/2 scram exists and thus IRM 11 cannot be INOP. The normal corrective action for an INOP neutron monitor is to bypass the monitor. Answer A is incorrect. The normal plant response from an upscale or inop IRM (in STARTUP mode) is both a rodblock and a 1/2 scram. But, it also shows that the HI-HI, HIGH, & DNSCL OR INOP lights are lit on IRM 13. The only way these can be lit is if IRM 13 were bypassed. With the IRM bypassed, it cannot generate any protective functions even though its count reading changes. Answers B & C are incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0029 LO 215-10444	

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**5**	**55.43**	

	(SRO Only)
	Time to Complete: 1-2 minutes

16 **ID: 09-1 NRO16** **Points: 1.00**

The plant is at rated power with the following air system indications:

- Air Compressor 1 is running in Lead
- Air Compressor 2 is in standby in Lag

Which of the following states the impact on Air Compressor 1 and Air Compressor 2 if **all** closed cooling water was lost to Air Compressor 1?

	Air Compressor 1	Air Compressor 2
A.	Will remain running	Remains in Standby
B.	Will auto trip from a high system air temperature signal	Will auto start on a low system air pressure signal
C.	Will auto trip from a high system air temperature signal	Will auto start immediately
D.	Will auto trip from a low cooling water pressure signal	Will auto start on a low system air pressure signal

Answer: B

Answer Explanation:

QID: 09-1 NRO16		
Question # / Answer	**16**	**Developer/Date: NTP 11/17/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

300000 Instrument Air A3.02 - Ability to monitor automatic operations of the INSTRUMENT AIR SYSTEM including: Air temperature				2.9	2.7
Level	**RO**	**Tier**	**2**	**Group**	**1**
General References	**RAP-M5a**	**334**			
Explanation	The plant is at rated power with the lead air compressor running and the lag air compressor in a standby condition. Air system pressure is normal. A loss of closed cooling water occurs to air compressor 1. The lead compressor (#1) will auto trip from a high air temperature signal, and the lead compressor will auto start when system air pressure lowers to the auto start setpoint. Answer B is correct. Answer A is incorrect since air compressor 1 trips and air compressor will auto start. Answer C is incorrect since the lag compressor will not start immediately since air pressure is above the auto start setpoint (95 psig). Answer D is incorrect since there is no compressor trip signal from low cooling water pressure.				
References to be provided during exam:	**None**				
Learning Objective	2621.828.0.0043 LO 279-10444				

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solving a problem with knowledge and its meaning			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

17 **ID: 09-1 NRO17** **Points: 1.00**

The plant was at rated power when an event occurred. The Operator reports that RPV water level is 80" and rising slowly.

One minute later, the following annunciator alarmed:

- COND B FLOW HI POSSIBLE RUPTURE

With **no** Operator action, which of the following states the final positions of the Isolation Condenser System Valves (Steam and Condensate Return) and associated Vent Valves?

	IC A	IC B
A.	System A Valves open Vent Valves open	System B Valves closed Vent Valves closed
B.	System A Valves open Vent Valves closed	System B Valves closed Vent Valves closed
C.	System A Valves open Vent Valves open	System B Valves open Vent Valves closed
D.	System A Valves open Vent Valves closed	System B Valves closed Vent Valves open

Answer: B

Answer Explanation:

QID: 09-1 NRO17		
Question # / Answer	**17**	**Developer/Date: NTP 12/11/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

207000 Isolation (Emergency) Condenser A3.03 - Ability to monitor automatic operations of the ISOLATION (EMERGENCY) CONDENSER including: Reactor water level					**3.5**	**3.7**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**EMG-SP1**		**307**		**RAP-C3b RAP-C1a**	
Explanation	The plant was at rated power when an event resulted in an RPV water level of 80". As water level lowers from its normal 155", both Isolation Condensers will auto initiate at 90" (1 condensate valve in each loop goes open to initiate the condenses, and the normally open vent valves go closed). So far, all System valves are open and all vent valves go closed. When the Possible Rupture B comes in, this will close all System valves in System B only, and the vent valves remain closed. Thus, System A valves are open with their associated vent valves closed. System B valves are closed with the vent valves closed. Answer B is correct. The other answers provide various positions. If the candidate does not know the initiation signals and their impact on the system/vent valves, or does not know how the Possible Rupture annunciator impacts the Isolation Condensers, than all other answers are plausible.					
References to be provided during exam:	**None**					
Learning Objective	2621.828.0.0023 LO 2030					

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

18 **ID: 09-1 NRO18** **Points: 1.00**

The plant was at rated power when the Operator reported the following observation:

- The Standby Liquid Control (SLC) System 1 continuity meter indicates 0 amps

Investigation revealed that the SLC Pumps were **not** impacted and that the squib valves remained physically intact.

An event then occurred which required SLC to be injected. The Operator placed the STANDBY LIQUID CONTROL keylock switch to the FIRE SYS 1 position.

Which of the following states the correct current panel indications and the next required Operator action?

	Indications	Required Action
A.	• SLC System 1 PUMP ON light is **lit** • Reactor power indicates lowering	• Verify PUMP DISCH PRESS greater than RPV pressure
B.	• STDBY LIC CNTRL - FLOW ON annunciator is **on** • Reactor power indicates lowering	• Verify RWCU isolated
C.	• SLC System 1 PUMP ON light is **lit** • Reactor power indicates stable	• Initiate SLC System 2
D.	• STDBY LIC CNTRL - FLOW ON annunciator is **on** • Reactor power indicates stable	• Initiate SLC System 2

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer: C

Answer Explanation:

QID: 09-1 NRO18		
Question # / Answer	18	Developer/Date: NTP 12/12/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
211000 SLC A2.02 - Ability to (a) predict the impacts of the following on the STANDBY LIQUID CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure of explosive valve to fire					3.6	3.9
Level	RO	Tier	2	Group	1	
General References	157B6350 sh. 188	RAP-G2b		EMG-SP22		

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant was at rated power when the SLC System 1 continuity meter indicated 0 amps. This indicates a loss of power to SLC System 1 explosive valve. When the SLC keylock is placed in FIRE SYS 1, this would normally start SLC Pump 1 and fire the associated explosive valve. In the case in the question, the pump will still start but the explosive valve will not fire and open. The SLC system explosive valve remains in its normal standby state when the keylock is placed in position 1. Because the pump starts, its indicated pressure will indicate it normal pressure of being > RPV pressure. Since the explosive valve will not open and the System 2 explosive valve remains closed, there will be no change in reactor power from this attempted SLC injection. Also, the FLOW ON annunciator will not be on since there is no SLC flow. IAW SP-22, if proper SLC indications are not received, then initiate SLC System 2. Answer C is correct. Answer A & B are incorrect since reactor power is not lowering. Answer D is incorrect since the Flow On annunciator is not lit.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0046 LO 211-10445	

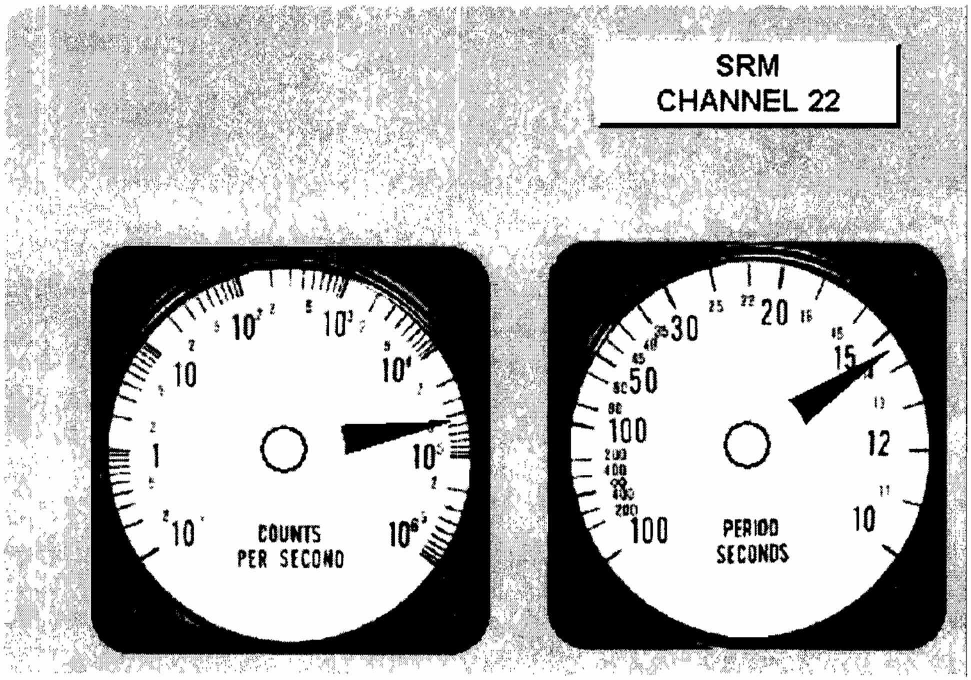
Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: predict an event or outcome			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

19 **ID: 09-1 NRO19** **Points: 1.00**

The plant is starting up after an outage and control rod withdrawals are in progress. As the Panel Operator is withdrawing a control rod in close proximity to SRM 22, another Operators reports the following observations:

- SRM Drawer 22 indications as shown
- All other SRMs Drawers show no changes

Which of the following states the correct RAPs that the Panel Operator will refer to now?



A. • SRM PERIOD SHORT **only**

B. • SRM PERIOD SHORT **and**
• ROD BLOCK **only**

C. • SRM HI/INOP **and**
• ROD BLOCK **and**
SCRAM CONTACTOR OPEN

D. • ROD BLOCK **and**
• SRM HI/INOP **only**

Answer: A

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer Explanation:

QID: 09-1 NRO19		
Question # / Answer	**19**	**Developer/Date: NTP 12/12/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
215004 Source Range Monitor A4.06 - Ability to manually operate and/or monitor in the control room: Alarms and lights						**3.2**	**3.1**
Level	**RO**	**Tier**	**2**	**Group**	**1**		
General References							
Explanation	The plant is starting up and withdrawing control rods. During a control rod withdrawal, an operators provides SRM 22 drawer indications, which include of note: SRM counts is on-scale, period indicates about 13 seconds and the amber PERIOD light is on. With these last 2 indications, then the SRM PERIOD SHORT annunciator will be in alarm (setpoint is 30 seconds). A fast period does not cause a rodblock, and current SRM counts does not show upscale or downscale. Therefore, the only annunciator that is in the alarm state is SRM PERIOD SHORT. Answer A is correct. Answer B is incorrect since a fast period does not give a rodblock signal, although SRM counts will. But the current SRM 22 counts is below that setpoint. Answer C is also incorrect since even an upscale SRM will not result in a 1/2 scram which gives the scram contactor open annunciator. Answer D is incorrect since there are no indications to support either a high SRM counts or inoperable condition.						
References to be provided during exam:	**None**						
Learning Objective	2621.828.0.0029 LO 215-10444						

Question Source (New, Modified, Bank)	**New**

Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:PEO
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

20 **ID: 09-1 NRO20** **Points: 1.00**

The plant is at 30% power on a startup. Current plant conditions include the following:

- The MASTER FEEDWATER LEVEL CONTROLLER is in AUTO
- Feedwater Pumps A and C are in service
- A MFRV FLOW CONTROLLER is in AUTO
- C MFRV FLOW CONTROLLER is in MAN, to be placed in AUTO

IAW 317, Feedwater System, which of the following displays on the C MFRV FLOW CONTROLLER should be about equal to each other, in order to place the C MFRV FLOW CONTROLLER in AUTO? .

A. P-display and the Y-display.

B. S-display and the V-display.

C. Y-display and the S-display.

D. V-display and the P-display.

Answer: B

Answer Explanation:

QID: 09-1 NRO20		
Question # / Answer	**20**	**Developer/Date: NTP 12/14/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
259002 Reactor Water Level Control A4.03 - Ability to manually operate and/or monitor in the control room: All individual component controllers when transferring from manual to automatic modes					**3.8**	**3.6**
Level	**RO**	**Tier**	**2**	**Group**	**1**	
General References	**317**					

Explanation	The plant is starting with 2 Feedwater Pumps in service, one in AUTO and the other in MAN. The Operator is ready to place the second controller in AUTO. To do this, the S-display and V-display must be made about equal to each other to prevent any changes in feedwater flow when the individual controller is placed on the master controller. When S and V are approximately matched, then the individual controller is placed in AUTO. Answer D is correct. The MFRV controller has displays for S, V, P and Y and the distracters represent the incorrect combinations of the displays.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0018 LO 259-10446	

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

21 **ID: 09-1 NRO21** **Points: 1.00**

The plant was at rated power when a loss of USS 1A2 occurred. While investigating the loss of the bus, the operating CRD pump tripped on overload.

The Operator manually scrammed the reactor and then reported the following indications:

- RPS Group 1 Scram lights on Panel 4F and 6R are **energized**
- Many LPRMs indicate > 2%
- Turbine Bypass Valves are controlling RPV pressure

IAW SP-21, Alternate Insertion of Control Rods, which of the following actions is required?

A. De-energize the Scram Solenoids by placing both 100 amp Main RPS Breakers in OFF.

B. Place the REACTOR MODE SELECTOR switch in REFUEL and manually insert control rods.

C. De-energize the Scram Solenoids by placing the Sub Channel Test Keylocks in the TRIP position.

D. Raise the CRD Cooling Water differential pressure by opening the CRD Cooling Water PCV NC40.

Answer: C

Answer Explanation:

QID: 09-1 NRO21		
Question # / Answer	**21**	**Developer/Date: NTP 12/14/09**

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
215005 APRM / LPRM 2.4.6 - Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.				3.7	4.7
Level	RO	Tier	2	Group	1

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

General References	EMG-SP21		
Explanation	The plant was at rated power when USS 1A2 was lost. This will remove power to CRD Pump A. When the second CRD Pump is lost on overload, all CRD pumps are lost and the operator scrams the plant. Indications show that several RPS Group solenoid lights are still energized and power is > 2% on several LPRMs. This shows that the plant is in an electrical ATWS. Also, with the turbine bypass valves controlling RPV pressure, the MSIVs must be open. IAW SP-21, for an electrical ATWS, a method to insert control rods is to place the sub-channel keylocks in TEST. Answer C is correct. The method in answer A can be used in an electrical ATWS but only when the MSIVs are closed. Answer A is incorrect. Because both CRD pumps are lost, the operator is unable to manually insert control rods. Answer B is incorrect. The method in answer D can be used in an electrical ATWS but with no CRD pumps, this will have no impact. Answer D is incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.845.0.0053 LO 200-10445A		

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPK
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

22 **ID: 09-1 NRO22** **Points: 1.00**

The plant was at rated power when an event occurred and the plant scrammed. The following observations and annunciators in alarm include:

- LKOUT RELAY 86/S1A TRIP
- LKOUT RELAY 86/S1B TRIP
- EDG 2 - LKOUT RELAY TRIP
- ATWOS RX RECIRC PUMP TRIP - ACTUATE A I, ACTUATE B I, ACTUATE C II **and** ACTUATE D II
- The RPV has been rapidly depressurized to 450 psig and lowering, due to an RPV leak in the Turbine Building

RPV water level continues to lower. Which of the following RPV water level instruments can be used to determine RPV water level?

A. GEMAC A

B. YARWAY B

C. Fuel Zone A

D. Fuel Zone B

Answer: C

Answer Explanation:

QID: 09-1 NRO22		
Question # / Answer	**22**	**Developer/Date: NTP 12/14/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
264000 EDGs 2.4.3 - Emergency Procedures / Plan: Ability to identify post-accident instrumentation.						**3.7**	**3.9**
Level	**RO**	**Tier**	**2**	**Group**	**1**		
General References	**EOP Users Guide**	**ABN-59 RAP-E1a**		**3013 sh. 1 RAP-E2a**			

Explanation	The plant was at rated power when an event occurred, including the following: the plant scrammed with a loss of offsite power, and the loss of EDG 2 to power Bus 1D. EDG 1 has started and is supplying Bus 1C, and the busses downstream: USS 1A2 & 1A3. USS 1A2 supplies VMCC 1A2, which supplies PAIPP-1. PAIPP-1 supplies power to Fuel Zone A. Answer C is correct. The question also shows that the recirculation pumps have tripped (ATWS annunciators) on RPV water level lo-lo of 90" and level is still lowering. The GEMAC lowest reading is 90" and is unable to provide indication less than 90". Answer A is incorrect. Because the RPV has rapidly depressurized to below 500 psig, and IAW the EOP Users Guide, the YARWAYS are not to be used to determine RPV water level. Answer B is incorrect. Answer D is incorrect since Fuel Zone 2 is powered from PAIPPS-2 (powered from VMCC 1B2, powered from USS 1B2, powered from Bus 1D) and has no power.
References to be provided during exam:	**None**
Learning Objective	2621.845.0.0052 LO 200-10445

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPR**
	NUREG 1021 Appendix B: Solve a problem using a reference			
10CRF55 Content	**55.41**	**6**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

23 **ID: 09-1 NRO23** **Points: 1.00**

Which of the following states the correct power supply to the listed Nuclear Instrument?

	Nuclear Instrument	Power Supply
A.	SRM 23	24 VDC Panel A
B.	SRM 24	24 VDC Panel B
C.	IRM 14	24 VDC Panel B
D.	IRM 15	24 VDC Panel A

Answer: B

Answer Explanation:

QID: 09-1 NRO23		
Question # / Answer	**23**	**Developer/Date: NTP 12/14/09**

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
215004 Source Range Monitor K2.01 - Knowledge of electrical power supplies to the following: SRM channels/detectors					2.6	2.8
Level	RO	Tier	2	Group	1	
General References	706E812 sh. 4	401.1				
Explanation	IAW the references, 24 VDC Panel B supplies SRMs 21 & 22, and IRMs 11-14. 24 VDC Panel B supplies SRMs 23 & 24, and IRMs 15-18. Answer B is correct. The other answers are plausible but incorrect.					

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

References to be provided during exam:	**None**	

Learning Objective	2621.828.0.0012 LO 263-10445

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:F**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Fact			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

24 **ID: 09-1 NRO24** **Points: 1.00**

The plant was at rated power when pressure switch PS-1A0083A, which inputs the RPV pressure signal into the open circuit for EMRV NR108A, failed **high**.

Which of the following states the required action regarding EMRV NR108A, and **following this action**, state the affect on EMRV NR108A to function for ADS?

	Required Action	Affect on ADS
A.	Place the AUTO DEPRESS VALVE NR108A switch in OFF	EMRV NR108A will **not** function for ADS
B.	Place the AUTO DEPRESS VALVE NR108A switch in OFF	ADS is unaffected
C.	Place the EMRV NR108A keylock switch in DISABLE	ADS is unaffected
D.	Place the EMRV NR108A keylock switch in DISABLE	EMRV NR108A will **not** function for ADS

Answer: B

Answer Explanation:

QID: 09-1 NRO24		
Question # / Answer	**24**	**Developer/Date: NTP 12/14/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

239002 SRVs **K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES : Nuclear boiler instrument system (pressure indication)**				**3.2**	**3.4**
Level	**RO**	**Tier**	**2**	**Group**	**1**

General References	**729E182 sh 1**	**ABN-40**	
Explanation	The plant is at power when the RPV pressure sensing pressure switch fails high to EMRV NR108A. This will result in EMRV NR108A only, opening. The crew will enter ABN-40, Stock Open EMRV. The first manipulation of EMRV controls is placing the control switch for the EMRV to OFF. This removes any RPV pressure input to the EMRV and it will close. With the switch in OFF, the ADS function is not impacted and the valve will open as designed for ADS. Answer B is correct. Answer A is incorrect since the ADS function is still operable. IAW the ABN, if the valve will not close using its control switch, then the DISABLE switch is used. This action will disable both the ADS function and the RPV pressure relief function. But, since the valve did close by placing its control switch in OFF, the DISABLE switch is not manipulated. Answers C & D are incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0005 LO 379		

Question Source (New, Modified, Bank)			**New**		
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**		**X** **3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning				
10CRF55 Content	**55.41**	**7**	**55.43**		
	(SRO Only)				
Time to Complete: 1-2 minutes					

25 **ID: 09-1 NRO25** **Points: 1.00**

The plant was at rated power when an ATWS occurred. The Operator placed the individual scram switch for control rod 02-27 to the scram position, and the control rod fully inserted.

Which of the following states the correct indications for control rod 02-27 with its scram switch in the scram position?

A. The closest LPRM 's amber light is energized and the control rod position displays 00 with green back-lighting.

B. The red scram light on Panel 4F is illuminated and the control rod position displays a blank with green back-lighting.

C. The SCRAM CONTACTOR OPEN annunciator is energized and the control rod position displays a blank with red back-lighting.

D. One of the RPS GROUP SCRAM lights is de-energized and the control rod position displays a blank with green back-lighting.

Answer: B

Answer Explanation:

QID: 09-1 NRO25		
Question # / Answer	**25**	**Developer/Date: NTP 12/15/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
212000 RPS A4.17 - Ability to manually operate and/or monitor in the control room: Perform alternate reactivity/ shutdown operations						**4.1**	**4.1**
Level	**RO**	**Tier**	**2**	**Group**	**1**		
General References	**Simulator**						

Explanation	The plant was at power when an ATWS occurred and the operator inserts control rods using the Rod Scram Test Panel (Panel 6XR). When a single switch is placed in the scram position, this will de-energize the scram solenoids for the single selected control rod and the rod will scram, as it would with a normal scram. The red scram light will energize, showing that the scram valves have opened, and with the scram signal still present, the control rod will indicate over-travel in the inward direction, which is a blank green-backlight (00 not indicated). The 00 position will show after the scram signal is removed. Answer B is correct. The closest LPRM will indicate lower as in answer A, but the rod will not show 00. Answer A is incorrect. Since the scram switch only de-energizes a single control rod scram solenoids, it does not affect a group scram solenoid light. Answer D is incorrect. When the control rod moves to the over-in position with the scram switch in the scram position, the rod position will not display the red backlight. Also, the scram contactor open annunciator does not alarm using the individual scram switches. Answer C is incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0011 LO 79	

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system status			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

26 **ID: 09-1 NRO26** **Points: 1.00**

The plant was at rated power with the following air compressor indications:

- Air Compressor 1 is running in Lead
- Air Compressor 2 is in Standby
- Air Compressor 3 is tagged out of service

The following annunciator then alarmed:

- COMPR 1 TRIP

Investigation revealed that the Air Compressor spuriously tripped on high bearing oil temperature as it had several times in the last few weeks. Bearing temperature was verified as normal. The SRO has directed a re-start of the compressor.

IAW 334, Instrument and Service Air, which of the following is correct to place Air Compressor 1 back in LEAD from the Control Room?

A. Reset the breaker locally. Then place the COMPRESSOR 1 control switch to START.

B. Reset the breaker locally. Then hold the COMPRESSOR 1 control switch in START for 3-5 seconds.

C. The local reset at the compressor must be pressed **once** and then wait for the start logic to be satisfied. Then place the COMPRESSOR 1 control switch to START.

D. The local reset at the compressor must be pressed **twice** and then wait for the start logic to be satisfied. Then hold the COMPRESSOR 1 control switch in START for 3-5 seconds.

Answer: D

Answer Explanation:

QID: 09-1 NRO26		
Question # / Answer	**26**	**Developer/Date: NTP 12/15/09**

Knowledge and Ability Reference Information

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

K&A	Importance Rating RO	Importance Rating SRO
300000 Instrument A K4.01 - Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Manual/automatic transfers of control	**2.8**	**2.9**

Level	**RO**	**Tier**	**2**	**Group**	**1**

General References	**334**	**RAP-M4a**	**RAP-M5a**
Explanation	The plant is at power with air compressor 1 is lead, air compressor 2 is standby, and air compressor 3 is tagged out of service. An annunciator alarms which describes a trip of compressor 1. The trip is spurious is and the SRO directs a restart of the compressor. IAW the procedure, to place air compressor 1 back in lead, the local reset at the compressor must be pressed twice and then wait for the start logic to be satisfied. Then hold the COMPRESSOR 1 control switch in START for 3-5 seconds. Answer D is correct. There is another annunciator, COMPR 1 BREAKER TRIP, which opens the air compressor breaker on overload only. This will require actions at the breaker to restart the air compressor. But this is not the annunciator given in the question. Answers A & B are incorrect. Answer C is incorrect since the air compressor switch must be held for 3-5 seconds, not just placed in start, and the reset must be pressed twice.		
References to be provided during exam:	**None**		
Learning Objective	2621.828.0.0043 LO 279-10447		

Question Source (New, Modified, Bank)				**New**
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

27 **ID: 09-1 NRO27** **Points: 1.00**

The plant is at rated power.

Complete the following statements:

(1) The loss of _____1_____ will result in the loss of the EPR.

(2) RPV pressure will stabilize at a _____2_____ value due to this power loss.

	(1)	(2)
A.	VACP-1	Lower
B.	VACP-1	Higher
C.	CIP-3	Lower
D.	CIP-3	Higher

Answer: D

Answer Explanation:

QID: 09-1 NRO27		
Question # / Answer	**27**	**Developer/Date: NTP 12/15/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
241000 Reactor/Turbine Pressure Regulator K1.14 - Knowledge of the physical connections and/or cause- effect relationships between REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following: A.C. electrical power	**2.8**	**2.9**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Level	RO	Tier	2	Group	2

General References	RAP-Q6a	ABN-58	315.4
Explanation	The plant is at rated power. In this condition, the EPR is in control, and the MPR relay position is 8-10% below that of the EPR. This means that the MPR setpoint is slightly above that of the EPR. When AC power (CIP-3) is lost to the EPR, the MPR takes control. Because the MPR setpoint was above the original EPR setpoint, it will call for a slightly lower turbine control valve opening which results in a slightly greater (than initial) RPV pressure. Therefore, if CIP-3 is lost, the MPR will control RPV pressure at a slightly higher pressure. Answer D is correct. Answers A & B are incorrect since they list an incorrect vital AC power supply. Answer C is incorrect since the event will result in a greater RPV pressure.		
References to be provided during exam:	None		
Learning Objective	2621.828.0.0051 LO 249-10446		

Question Source (New, Modified, Bank)				Modified
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:PEO
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

28 **ID: 09-1 NRO28** **Points: 1.00**

The plant was at rated power when a loss of offsite power event combined with a loss of coolant accident occurred.

The Operator then reports the following annunciators in alarm:

- 1A2 MN BRKR OL TRIP
- 1A2 MN BRKR TRIP

Which of the following Containment Spray Loops remain available for the Containment Spray function?

A. Loops A and B

B. Loops C and D

C. Loops A and C

D. Loops B and D

Answer: B

Answer Explanation:

QID: 09-1 NRO28		
Question # / Answer	**28**	**Developer/Date: NTP 12/15/09**

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
226001 RHR/LPCI: CTMT Spray Mode K2.02 - Knowledge of electrical power supplies to the following: Pumps				2.9	2.9
Level	RO	Tier	2	Group	2
General References	310		RAP-U4a		

Explanation	The plant was at power when a LOOP/LOCA occurred. Then, USS 1A2 was lost. Containment Spray pumps 51C and 51D are powered from 1B2 (which is currently being supplied from EDG 2). The Containment Spray System 2 valves needed to change state for the containment spray function are powered from 1B21B, which is fed from USS 1B2, which is currently being fed from EDG 2. Therefore, Pumps C & D are still available to perform the containment spray function. Answer B is correct. The other distracters list at least one pump which is not powered under the given circumstances.
References to be provided during exam:	None
Learning Objective	2624.828.0.0009 LO 226-10453

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

29 **ID: 09-1 NRO29** **Points: 1.00**

The plant was at 50% power with control rod withdrawals in progress. Control rod 38-23 is the next control rod to be withdrawn. Control rod 38-23 is currently at position 00 and is to be withdrawn to its final position of 12 in the current RWM Step.

Consider the following sequence of events:

- Control rod 38-23 became uncoupled and stuck at position 00
- The drive for control rod 38-23 was withdrawn to the correct position
- The next control rod was selected and was being withdrawn

If control rod 38-23 then became completely unstuck, which of the following states the **first** plant indication that control rod 38-23 is now unstuck?

A. A rise in LPRMs near the control rod.

B. The ROD DRIFT annunciator will alarm.

C. The ROD OVERTRAVEL annunciator will alarm.

D. A rapid change in indicated control rod 38-23 position.

Answer: A

Answer Explanation:

QID: 09-1 NRO29		
Question # / Answer	**29**	**Developer/Date: NTP 12/15/09**

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
201003 Control Rod and Drive Mechanism K3.01 - Knowledge of the effect that a loss or malfunction of the CONTROL ROD AND DRIVE MECHANISM will have on following: Reactor power					3.2	3.4
Level	RO	Tier	2	Group	2	
General References	UFSAR 15.4.9	205		2621.828.0.0029		

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant is at 50% power with control rod withdrawals in progress. Control rod 38-23 is about to be withdrawn from position 00 to its final position of 12 in the current RWM Step. The control rod becomes uncoupled and stuck. Later, the control rod becomes unstuck and falls (drops) out of the core. The first indication will be a rise in LPRMs/APRMs located next to the unstuck control rod. Answer A is correct. The Rod Drift alarm will not annunciate since there will be no change in rod position reed switches. Answer B is incorrect. An uncoupled rod can be diagnosed by the Overtravel annunciator but this is only performed at position 48 and the control rod only went to position 12. Answer C is incorrect. Answer D is incorrect since rod position is independent from the control rod blade itself (but is dependent on the drive piston, which has been withdrawn to position 12) and will show position 12 prior to the control rod drop. Answer D is incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0011 LO 201-10450	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

30 **ID: 09-1 NRO30** **Points: 1.00**

IAW ABN-6, Control Rod Malfunctions, which of the following events requires the Operator to place the NOTCH OVERRIDE switch to the EMERG ROD IN position?

A. When there are simultaneous multiple drifting control rods.

B. An outward drifting control rod when there is an RMCS timer malfunction.

C. A control rod is to be inserted following a double-notching event on a withdrawal.

D. When a withdrawn control rod will not insert with the ROD CONTROL switch in the ROD IN position.

Answer: B

Answer Explanation:

QID: 09-1 NRO30		
Question # / Answer	**30**	**Developer/Date: NTP 12/15/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
201002 RMCS **K4.06 - Knowledge of REACTOR MANUAL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Emergency In rod insertion**					**3.5**	**3.5**
Level	**RO**	**Tier**	**2**	**Group**	**2**	
General References	**ABN-6**					
Explanation	The reference states that if one control rod is moving out and a timer malfunction is indicated, then apply an EMERG ROD IN signal using the NOTCH OVERRIDE switch. Answer B is correct. All other distracters are incorrect but use of the EMERG ROD IN would function to insert a malfunctioning control rod and is plausible.					

References to be provided during exam:	None	
Learning Objective	2621.828.0.0011 LO 201-10450	

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge	X 1:P	Comprehension or Analysis	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

31 **ID: 09-1 NRO31** **Points: 1.00**

Consider the plant at rated power in 2 different conditions:

Condition 1: **No** annunciators alarmed
Condition 2: The following annunciators alarmed:

- TORUS/DW 1 VAC BRKR OPEN
- TORUS/DW 2 VAC BRKR OPEN

Which of the following is correct given the indications above?

During a LOCA in the Primary Containment, ...

A. indicated Torus pressure will **rise more rapidly** in **Condition 2**.

B. indicated Torus water level will **rise more rapidly** in **Condition 2**.

C. indicated Drywell pressure will **rise more slowly** in **Condition 1**.

D. the amount of water in the Downcomers will be **higher** in **Condition 1**.

Answer: A

Answer Explanation:

QID: 09-1 NRO31		
Question # / Answer	**31**	**Developer/Date: NTP 12/16/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
223001 Primary CTMT and Aux. K5.01 - Knowledge of the operational implications of the following concepts as they apply to PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES : Vacuum breaker/relief operation	**3.1**	**3.3**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Level	RO	Tier	2	Group	2
General References	GU 3E-243-21-1000 sh. 1	UFSAR 6.2.1.1.1 RAP-C4f, -C5f	604.4.016		

Explanation	There are 7 sets of vacuum breakers (of 2 parallel valves in each set) between the Drywell and the Torus air space. The valves are normally closed and automatically open when Drywell pressure drops below Torus pressure by about 0.5 psig. This helps to prevent exceeding the Drywell design negative pressure. Condition 2 shows that at least one vacuum breaker is open. This allows direct communication between the Drywell and the Torus air space. When a LOCA occurs in the Drywell, steam is designed to flow down the downcomers to below the Torus water level to suppress the steam. With a vacuum breaker open, some steam can flow directly from the Drywell into the Torus air space and this steam is not condensed and its pressure is not suppressed. Thus during a LOCA, the indicated torus pressure will rise faster in Condition 2. Answer A is correct. With additional pressure above the water level in the Torus, this will push more water up into the downcomers. Thus water level will not be higher for the same size LOCA in Condition 2. Answer B is incorrect. Because the steam from the LOCA can expand into the Torus space in Condition 2, Drywell pressure will rise more slowly in Condition 2 - not Condition 1. Answer C is incorrect. In LOCA conditions, Torus pressure lags behind Drywell pressure as Drywell pressure rises. With the vacuum breaker open, Torus pressure will be closer to Drywell pressure. As a result, the Dp between the Drywell and Torus will be smaller in Condition 2. With a smaller Dp, there will be more water in the downcomer in Condition 2. Answer D is incorrect.

References to be provided during exam:	None	
Learning Objective	2621.828.0.0032 LO 432	

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge	X 1:F	Comprehension or Analysis	

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

	NUREG 1021 Appendix B: Fact				
	10CRF55 Content	55.41	5	55.43	
		(SRO Only)			
	Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

32 **ID: 09-1 NRO32** **Points: 1.00**

The plant was at 50% power during a startup with control rod manipulations in progress. The ROD WORTH MINIMIZER switch is in NORMAL.

An event then occurred which resulted in the loss of power to the RWM and the Plant Computer System.

Which of the following states the impact of the power supply loss on the Rod Worth Minimizer?

A. The RWM **will insert** a control rod block due to the loss of power.

B. The RWM **will not** insert a rod block from the power loss **but** will **not** enforce the control rod pattern.

C. The RWM **will not** insert a control rod block since reactor power is above the Low Power Setpoint.

D. The RWM **will insert** a control rod block due to the loss of control rod position information from the PPC.

Answer: A

Answer Explanation:

QID: 09-1 NRO32		
Question # / Answer	**32**	**Developer/Date: NTP 12/16/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
201006 RWM K6.01 - Knowledge of the effect that a loss or malfunction of the following will have on the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC): Power supply				**2.8**	**3.2**
Level	**RO**	**Tier**	**2**	**Group**	**2**
General References	**2621.828.0.0041**	**409**		**201**	

Explanation	The plant is starting up with control rod withdrawals in progress. At this point in the startup, the RWM is in the Power Operations Mode, which acts very similarly to the low power mode in that it will still insert control rod blocks if control rods manipulations deviate from the planned control rod manipulations. The PPC UPS also powers the RWM. With the RWM in service (ie, not bypassed, or above the low power setpoint while in the low power mode), a loss of power to the RWM will result in a control rod block. Answer A is incorrect. The RWM sends control rod information to the PPC. Since the power loss does result in a rod block, answer B is incorrect. It is true that the RWM is above the LPSP, but the RWM will be in the power operations mode at this time in the startup and the LPSP does not apply, and a rodblock will still be applied from the loss of power. Answer C is incorrect. The RWM supplies control rod information to the PPC - not the other way around. Thus, there is a rodblock but not because the PPC does not/cannot provide control rod position information to the RWM. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2624.828.0.0041 LO 217-10444	

Question Source (New, Modified, Bank)			**New**		
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**	
	NUREG 1021 Appendix B: Predict an event or outcome				
10CRF55 Content	**55.41**		**55.43**		
	(SRO Only)				
Time to Complete: 1-2 minutes					

33 **ID: 09-1 NRO33** **Points: 1.00**

The Spent Fuel Pool Cooling System was in service with one pump, filter and demineralizer. The flow controller is currently set at 70% open with flow established at 400 GPM when the following occurred:

- The flow controller failed to 100% open

Which of the following states the **initial** affect on water level in the Skimmer Surge Tanks **and** in the Fuel Pool?

	Skimmer Surge Tanks Level	Fuel Pool Level
A.	Lower	Lower
B.	Higher	Higher
C.	Lower	Higher
D.	Higher	Lower

Answer: C

Answer Explanation:

QID: 09-1 NRO33		
Question # / Answer	33	Developer/Date: NTP 12/17/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
233000 Fuel Pool Cooling/Cleanup A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the FUEL POOL COOLING AND CLEAN-UP controls including: Surge tank level					2.6	2.9
Level	RO	Tier		Group		

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

General References	237E756		
Explanation	The fuel pool overflows into the skimmer surge tanks. The fuel pool cooling pumps take a suction on the skimmer surge tanks and discharge directly back into the fuel pool. The fuel pool water level and skimmer surge tanks water level is in equilibrium and steady at a system flow rate of 400 gpm. To maintain steady levels, 400 gpm must flow from the fuel pool into the skimmer surge tanks. When the flow controller is opened further, the skimmer surge tank level will drop to accommodate more flow. As the flow rate rises into the fuel pool, water level there will initially rise, until again equilibrium is achieved. Answer C is correct. The other answers are plausible if the candidate does not understand the system and flow path.		
References to be provided during exam:	None		
Learning Objective	2621.828.0.0020 LO 231-10445		

Question Source (New, Modified, Bank)			Bank	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:PEO
	NUREG 1021 Appendix B:			
10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

34 **ID: 09-1 NRO34** **Points: 1.00**

The plant was at rated power. The Operator had just placed TIP 3 and 4 at the core top location, when the following annunciators alarmed:

- CORE SPRAY - SYSTEM 1 AUTOSTART
- CORE SPRAY - SYSTEM 2 AUTOSTART

10 minutes later, the Operator reports the following observations:

- TIP CHANNEL 3
  - IN SHIELD white light is energized
  - DETECTOR POSITION displays 02
- TIP CHANNEL 4
  - IN SHIELD white light is de-energized
  - DETECTOR POSITION displays 255
- The TIP red light (Panel 11F) is energized
- **No** TIPs can be moved

Which of the following states the status of the TIPs 3 & 4, and the required actions IAW 405.2, Operation of the TIP System?

	TIP Status	Required Action
A.	TIP 3 has isolated TIP 4 has **not** isolated	Fire the shear valve for TIP 4
B.	TIP 4 has isolated TIP 3 has **not** isolated	Fire the shear valve for TIP 3
C.	TIP 3 has isolated TIP 4 has **not** isolated	Manually retract TIP 4 locally
D.	TIP 4 has isolated TIP 3 has **not** isolated	Manually retract TIP 3 locally

Answer: A

Answer Explanation:

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

QID: 09-1 NRO34		
Question # / Answer	34	Developer/Date: NTP 12/17/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
215001 Traversing In-core Probe A2.07 - Ability to (a) predict the impacts of the following on the TRAVERSING IN-CORE PROBE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Failure to retract during accident conditions: Mark-I&II(Not-BWR1)					3.4	3.7
Level	RO	Tier	2	Group	2	
General References	405.2		RAP-B1e RAP-B1f		EMG-SP1	

Explanation	The plant is at power with TIPs 3 & 4 at the core top. The provided annunciators show that a LOCA signal has been generated (3 psig Drywell pressure or RPV water level at or below 86"). These signals also isolate the Primary Containment and RPV, including the TIPs. On an isolation, the TIPs automatically retract and the ball valves close. Conditions show that with the Panel 11F TIP red light on, then at least one TIP ball valve is open. It also shows that the in shield light for TIP 4 is de-energized, which means that the TIP 4 has not retracted to the in shield position and the ball valve will be open. The ball valve normally auto closes when the TIP is retracted into the shield. The TIP 4 detector position (lowest is in shield and counts up as the detector moves out of the shield) shows that it is not in shield. IAW the 405.2, with a ball valve open and cannot be closed, then it directs that the shear valve be fired for the applicable TIP. Answer A is correct. Answer B is incorrect since it lists the incorrect TIP as not being in-shield. Answer C is incorrect since it lists the correct TIP but the incorrect action, although TIPs can be manually cranked locally . Answer C is incorrect. Answer D is incorrect since although TIPs can be manually cranked locally, this is not the procedure direction. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0029 LO 215-10445	

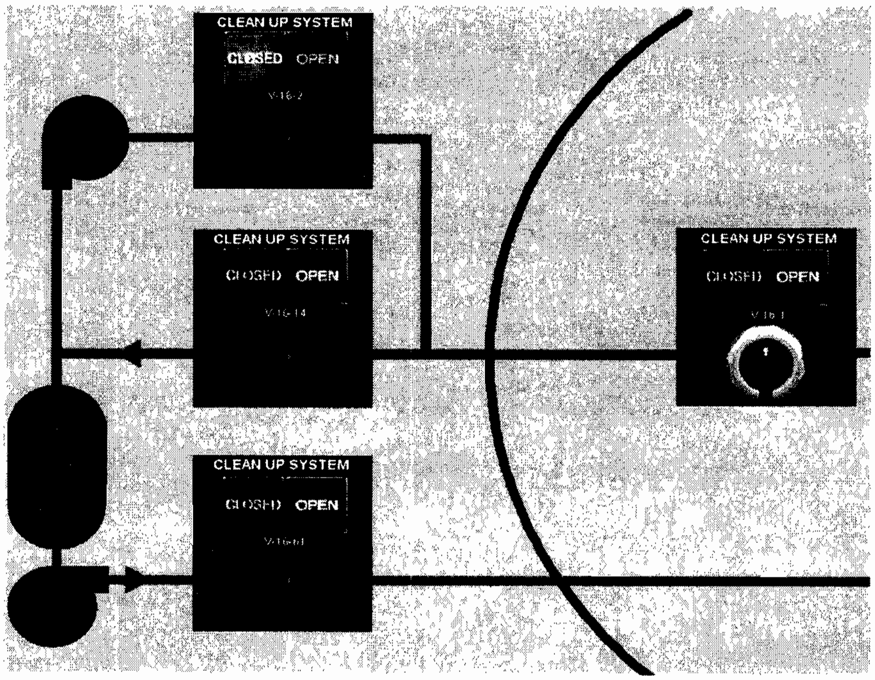
Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

35 **ID: 09-1 NRO35** **Points: 1.00**

The plant was at rated power when an event occurred, which resulted in the following annunciator:

- NRHX OUTLET TEMP HI

Which of the following states **all** RWCU valves that receive an isolation signal?



A. V-16-1
V-16-14
V-16-61

B. V-16-1
V-16-2
V-16-14

C. V-16-1
V-16-2
V-16-61

D. V-16-1
V-16-2
V-16-14
V-16-61

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer: B

Answer Explanation:

QID: 09-1 NRO35		
Question # / Answer	35	Developer/Date: NTP 12/17/09

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
204000 RWCU A3.05 - Ability to monitor automatic operations of the REACTOR WATER CLEANUP SYSTEM including: Reactor water temperature				2.8	2.8
Level	RO	Tier	2	Group	2
General References	RAP-D8b		EMG-SP1	148F444, sh. 1	
Explanation	The plant was at power when an event occurred which resulted in the alarm provided. When this alarms activates, it signals that V-16-1, V-16-2 and V-126-14 will auto close. Answer B is correct. All 4 valves will auto close on a Drywell high pressure, a RPV lo-lo water level, or a sensed RWCY line break. All other answers are plausible but incorrect.				
References to be provided during exam:	None				
Learning Objective	2621.828.0.0039 LO 204-10444				

Question Source (New, Modified, Bank)			Bank	
Cognitive Level	Memory or Fundamental Knowledge	X 1:I	Comprehension or Analysis	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	55.41		55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

36 **ID: 09-1 NRO36** **Points: 1.00**

The plant was at rated power with CRD Pump A in service, when a manual scram was inserted. The Operator reported the following indications at the time of the turbine trip:

- STARTUP BREAKER S1A indicated green light **on** and will not close
- STARTUP BREAKER S1B indicated green light **on** and will not close
- Annunciator MN BRKR 1C 86 LKOUT TRIP is in alarm

Which of the following states the status of the CRD Pump breakers **2 minutes** after the above indications?

	CRD Pump A	CRD Pump B
A.	Green light on	Green light on
B.	Red light on	Red light on
C.	Red light on	Green light on
D.	Green light on	Red light on

Answer: D

Answer Explanation:

QID: 09-1 NRO36		
Question # / Answer	**36**	**Developer/Date: NTP 12/17/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
201001 CRD Hydraulic A4.01 - Ability to manually operate and/or monitor in the control room: CRD pumps				**3.1**	**3.1**
Level	**RO**	**Tier**	**2**	**Group**	**2**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

General References	**341**	**RAP-T2a**	
Explanation	The plant is at power with CRD Pump A running, when a manual scram was inserted. When the turbine tripped, indications show a loss of offsite power and 4160 VAC Bus 1C has a fault (86 lockout). Under a loss of offsite power condition, with no indications of a LOCA, both EDGs will fast start and close onto Bus 1C and 1D. 60 seconds after the busses are re-energized, a CRD Pump will auto start on USS 1A2 (EDG 1 feed) and USS 1B2 (EDG 2 feed). But, since Bus 1C has an 86 lockout, EDG 1 will not fast start and load onto it emergency bus, and the downstream busses remain de-energized. Thus, CRD Pump A is de-energized (green light on) and CRD Pump B re-energizes (red light on). Answer D is correct. If the candidate confuses the power supplies to the pumps, or mis-applies the lockout condition, any other answer will seem plausible.		
References to be provided during exam:	**None**		
Learning Objective	2624.828.0.0012 LO 263-10453		

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

37 **ID: 09-1 NRO37** **Points: 1.00**

The plant was at rated power when the Shift Manager declared that a Control Room evacuation was required, and ABN-30, Control Room Evacuation, was entered.

The Operator only had enough time to scram the reactor and verify all control rods inserted, when **all** operators left the Control Room.

IAW ABN-30, which of the following states the required action outside of the Control Room and the associated plant impact of the action?

	Required Action	Plant Impact
A.	Close the MSIVs by isolating the air supply locally	The use of Isolation Condenser B for RPV pressure control at the RSP may be required
B.	Trip and lockout 4160 VAC 1D Main Breaker	Will require starting EDG 2 at the LSP DG2
C.	Trip all Condensate Transfer Pump breakers locally	May require controlling Condensate Transfer Pump at LSP 1B32
D.	Trip and lockout all Feedwater Pump breakers	CRD injection with CRD Pump B at the RSP may be required

Answer: D

Answer Explanation:

QID: 09-1 NRO37		
Question # / Answer	**37**	**Developer/Date: NTP 12/18/09**

Knowledge and Ability Reference Information	
K&A	**Importance Rating**

	RO	SRO
259001 Reactor Feedwater **2.4.34 - Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.**	**4.2**	**4.1**

Level	**RO**	**Tier**	**2**	**Group**	**2**

General References	**ABN-30**		
Explanation	The plant was at rated power when a control room evacuation was required and ABN-30 was entered. The following actions are attempted, if possible, prior to leaving the control room: Critical steps include: scram, trip recirculation pumps, close MSIVs, and trip all feedwater pumps. There are many other non-critical steps as well. If these critical steps cannot be performed prior to leaving the control room, Attachment ABN-30-1 provides direction on how to perform these actions in the plant. From what is given, the reactor was scrammed prior to leaving the control. The ABN does require tripping the feedwater pumps at the breakers. With a loss of high pressure injection now gone, using CRD Pump B from the RSP may be required. Answer D is correct. Closing the MSIVs is a critical step in the ABN, but the listed method to close the MSIVs is incorrect. The plant impact is also correct. Answer A is incorrect. Trip and lockout of breaker 1D is mentioned in the ABN, but is only required if no offsite power is available. The conditions stipulate that a successful scram was performed and can be assumed that offsite power is available. Therefore, answer B is incorrect. The ABN does not require tripping of condensate transfer pumps although it does require tripping of condensate pumps (non-critical step). Controlling the condensate transfer pump at the LSP is correct. But, answer D is incorrect since it refers to condensate transfer pumps and not condensate.		

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0017 LO 259-10445	

Question Source (New, Modified, Bank)	**New**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Cognitive Level	Memory or Fundamental Knowledge	X 1:P	Comprehension or Analysis	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

38 **ID: 09-1 NRO38** **Points: 1.00**

The plant was at 95% power. The Operator had just completed control rod withdrawals IAW the ReMA. The PPC Reactor Core State Parameters **prior to** and 15 minutes **after** the control rod manipulations are shown below:

Prior to control rod manipulations:

Thermal Limit	Fraction of Limit
MFLCPR	0.899
MFLPD	0.910
MAPRAT	0.917

After control rod manipulations:

Thermal Limit	Fraction of Limit
MFLCPR	0.995
MFLPD	1.002
MAPRAT	0.976

Which of the following states the potential impact on the nuclear fuel, and the **most restrictive** required action under the given conditions, IAW 202.1, Power Operations?

	Fuel Impact	Required Action
A.	Failures from transition boiling	Contact US, RE and monitor trend
B.	Failures from transition boiling	Follow TS 3.10, Notify RE Manager and Director Operations
C.	Failures from fuel pellet expansion	Follow TS 3.10, Notify RE Manager and Director Operations
D.	Failures from fuel pellet expansion	Contact US, RE, and monitor trend

Answer: C

Answer Explanation:

QID: 09-1 NRO38		
Question # / Answer	38	Developer/Date: NTP 12/18/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
290002 Reactor Vessel Internals A2.05 - Ability to (a) predict the impacts of the following on the REACTOR VESSEL INTERNALS ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Exceeding thermal limits					3.7	4.2
Level	RO	Tier	2	Group	2	
General References	202.1	GFES Thermo Chapter 9				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant is at 95% power with control rod withdrawals just completed. The before/after reactor core state parameters data is provided. The before data shows that all thermal limits are within allowable limits. The after data shows that MFLPD is > 1.00. This would require the action listed in Column C of Attachment 202.1-5 and is the most restrictive action. Is can also be seen that MFLCPR is in violation of Column B of Attachment 202.1-5, but its action from the Attachment is less restrictive. IAW 202.1, Attachment 5, if MFLPD is > 0.98, then the action is to contact the US, RE and to monitor. When > 0.99, then restore to within the limits. When > 1.00, then TS 3.10 must be applied (which requires restoring back below the limit)rations. Operating the reactor with MFLPD > 1.00 can result in fuel failures due to fuel pellet expansion. Answer C is correct. Answers A is incorrect since these fuel failures can result from exceeding CPR limits, which are in violation of Column B in the Attachment, but the action in Column B is less restrictive than Column C in the correct answer. Answer B is incorrect since the transition boiling thermal limit MFLCPR does not violate the given action, which is the response when Column C is violated. Currently, MFLCPR only violates Column A & B. Answer D fuel impact may occur and shows that Column A limit for MAPRAT is exceeded , but the action is less limiting than the correct answer C. Answer D is incorrect.
References to be provided during exam:	**202.1 Attachment 5**
Learning Objective	2621.850.0.0090 LO 1520

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPR**
	NUREG 1021 Appendix B: Solve a problem with references			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

39 **ID: 09-1 NRO39** **Points: 1.00**

The plant was at rated power when an event occurred. The Operator reports the following observations:

- RPV pressure peaked at 1380 psig for 5 seconds, then quickly lowered to between 1000 - 1100 psig
- RPV water level lowered to 78" for 5 seconds then quickly restored to above 100"

Which of the following is correct regarding Safety Limits as defined in Tech Specs and the current plant status?

	Safety Limit	Plant Status
A.	• The Reactor Coolant System pressure safety Limit was exceeded	• The Isolation Condensers are in service • The Core Spray System has initiated
B.	• The Reactor Coolant System Pressure Safety Limit was exceeded	• The Isolation Condensers are in service • The Core Spray System has initiated and is injecting
C.	• The Fuel Cladding Integrity Safety Limit has been exceeded	• **All** EMRVs and SRVs have opened and are now closed
D.	• The Fuel Cladding Integrity Safety Limit has been exceeded	• The Core Spray System has initiated • Both EDGs have idle started

Answer: A

Answer Explanation:

QID: 09-1 NRO39

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Question # / Answer	39	Developer/Date: NTP 12/18/09

Knowledge and Ability Reference Information							
K&A						Importance Rating	
						RO	SRO
295025 High Reactor Pressure EK1.05 - Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR PRESSURE : Exceeding safety limits						4.4	4.7
Level	RO	Tier	1	Group	1		

General References	TS 2.1	TS 2.2	EMG-SP1
Explanation	The plant was at power when an event resulted in high RPV pressure and low RPV water level. The fuel cladding integrity safety limit for RPV water level is no less than 4'8" above the fuel, or 56". The RPV pressure safety limit is no greater than 1375 psig with fuel in the vessel. Therefore, the pressure safety limit has been exceeded. The isolation Condensers auto initiate on 1050 psig or RPV water level 90", and thus are in service. Core Spray auto initiates at 86" RPV water level or Drywell pressure of 3 psig. Thus, the Core Spray System is in service, although it will not inject until RPV pressure drops to 305 psig. Therefore, the RPV pressure safety limit has been exceeded and core spray has initiated and isolation condensers are in service. Answer A is correct. Because RPV pressure is greater than the core spray injection setpoint, core spray is not injecting. Answer B is incorrect. Both answers C & D are incorrect since the fuel clad integrity safety limit has not been exceeded. Both plant statuses are correct.		
References to be provided during exam:	None		
Learning Objective	2621.850.0.0090 LO 1658		

Question Source (New, Modified, Bank)	New

Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPK
	NUREG 1021 Appendix B: Solve a problem using			
10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

40 **ID: 09-1 NRO40** **Points: 1.00**

The plant was at rated power when an earthquake occurred. The Operator makes the following report:

- RPV water level is 80" and lowering slowly
- Breaker S1B indicates open
- A fire alarm has alarmed on Main Fire Panel A

A report is received of a fire in the RB NW Corner Room and the Fire Brigade is dispatched. The Fire Brigade Leader reports that the Fire Brigade actions will impact the operation of **all** pumps located in the NW corner room.

Which of the following states the RPV injection sources currently available and **not** impacted by the fire?

A.
- Feedwater Pumps B & C
- Core Spray Main Pumps A & B

B.
- Feedwater Pump A
- Core Spray Main Pumps B & D

C.
- Feedwater Pump A & B
- SLC Pumps A & B

D.
- Feedwater Pump A
- Core Spray Main Pumps B & C
- CRD Pumps A & B

Answer: B

Answer Explanation:

QID: 09-1 NRO40		
Question # / Answer	**40**	**Developer/Date: NTP 12/19/09**

Knowledge and Ability Reference Information	
K&A	**Importance Rating**

	RO	SRO
600000 Plant Fire On-site AK1.02 - Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire fighting	2.9	3.1

Level	RO	Tier	1	Group	1

General References	3E-153-02-001	BR 3001B	
Explanation	The plant was at rated power when an earthquake occurred. This resulted in a low RPV water level, the loss of startup transformer 1B and a fire in the plant. The fire has been confirmed to impact all equipment in the RB NW corner room. This room contains both CRD pumps, and Core Spray main pumps A & C. Thus, this equipment is not available for RPV injection. Feedwater Pump A and Core Sprays B & D are available, both electrically and not impacted by the fire. Answer B is correct. Answer A is incorrect since Feedwater Pumps B & C are powered from Bus 1B, which has no power. Answer C is incorrect since it lists Feedwater Pump B. Answer D is incorrect since it lists the incorrect Core Spray pumps and it lists CRD pumps.		
References to be provided during exam:	None		
Learning Objective	2621.828.0.0010 LO 209-10445		

Question Source (New, Modified, Bank)				New
Cognitive Level	Memory or Fundamental Knowledge	X 1:S	Comprehension or Analysis	
	NUREG 1021 Appendix B: Structures and locations			
10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

41 **ID: 09-1 NRO41** **Points: 1.00**

The plant was shutdown and was cooling down with Shutdown Cooling (SDC) Pumps A and B, when the following annunciator alarmed:

- 1A2 DC LOST

Which of the following states the affect on Shutdown Cooling?

A. SDC Pump A is **only** able to be tripped from the LSP.

B. SDC Pump A discharge valve, V-17-55, **cannot** be closed.

C. SDC Pump A is **unable** to trip on high suction temperature.

D. The RPV cooldown rate can **not** be adjusted from the Control Room.

Answer: C

Answer Explanation:

QID: 09-1 NRO41		
Question # / Answer	41	Developer/Date: NTP 12/19/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
295004 Partial or Total Loss of DC Pwr AK1.05 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Loss of breaker protection					3.3	3.4
Level	RO	Tier	1	Group	1	
General References	BR E1129 157B6397 sh. 4	RAP-U3d		305		

Explanation	The plant was cooling down with SDC pumps A & B when DC power is indicated lost to USS Bus 1A2. USS 1A2 powers SDC Pump A. Therefore, DC control power is lost to SDC Pump A and not SDC Pump B. Since DC control power is lost, it cannot auto trip on high suction temperature. Answer C is correct. The same DC power is required for tripping when SDC Pump A is controlled from the LSP. Answer A is incorrect. SDC Pump A discharge valve, V-17-55, is a motor operated valve, but is powered from a DC source (DC-1, which is normally powered from DC-B) and is not impacted by the loss of the DC source to USS 1A2 (DC-C). Answer B is incorrect. The cooldown rate can still be adjusted from the control by several methods, one of which by tripping SDC Pump B. Answer D is incorrect. 4/13/10: This question has been modified due to NRC feedback.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0045 LO 205-10453	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:F**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Facts			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

42 **ID: 09-1 NRO42** **Points: 1.00**

The plant was at rated power when a leak developed in the Torus. A timeline of Torus water level is provided below (times are in minutes):

- T = 0 Torus water level indicated 145"
- T = 5 Torus water level indicated 140"
- T = 15 Torus water level indicated 130"

Which of the following represents the **soonest** that steam from a LOCA in the Primary Containment would **directly** pressurize the Torus air space?

A. T = 30 minutes

B. T = 36 minutes

C. T = 42 minutes

D. T = 56 minutes

Answer: B

Answer Explanation:

QID: 09-1 NRO42		
Question # / Answer	**42**	**Developer/Date: NTP 12/21/09**

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
295030 Low Suppression Pool Water Level EK2.07 - Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following: Downcomer/ horizontal vent submergence				3.5	3.8
Level	RO	Tier	1	Group	1
General References	EOP Users Guide				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant was at rated power when a Torus water leak developed. The indications show that Torus water level is lowering at the rate of 1"/minute. IAW the reference, below 110", the Drywell vent header downcomer openings are uncovered and the pressure suppression function of the Primary Containment becomes inoperable. Steam discharged from a LOCA would exit the downcomers, bypass the water in the Torus and directly pressurize the Torus air space. Thus at an additional 20 minutes (or T = 35), Torus water level will be 110", and at T = 36 minutes, Torus water level will be less than 110". Answer b is correct. At T=30 minutes, the downcomers are still covered. Answer A is incorrect. At T=42 minutes, Torus water level is even further below 110", but it is not the soonest time that the downcomers become uncovered. Answer C is incorrect. At T = 56 minutes, Torus water level has past 90", which is the level that the EMRV discharge pipes become uncovered. Answer D is incorrect. All other answers are plausible but incorrect.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0032 LO 432

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

43 **ID: 09-1 NRO43** **Points: 1.00**

The plant was at rated power when a total loss of TBCCW occurred. The Operators performed **all** IMMEDIATE OPERATOR ACTIONs of ABN-1, Reactor Scram **and** ABN-20, TBCCW Failure Response.

One minute later, the Operator reports the following:

- All control rods indicate full-in
- RPV water level indicates 120" and lowering slowly
- TOTAL FEEDWATER FLOW indicates 6 x 10^6 LBS/HR
- RPV pressure indicates 920 psig and lowering slowly

Which of the following is the **next** annunciator to alarm under the given conditions that will require manual Operator actions IAW ABN-20?

A. GENERATOR - MN LEADS TEMP HI

B. RX RECIRC PUMP TRIP - MG BRG TEMP HI

C. CLEANUP SYSTEM - AUX PUMP CCW TEMP HI

D. FEED PUMPS - COND/FW PMP BRG TEMP HI

Answer: D

Answer Explanation:

QID: 09-1 NRO43		
Question # / Answer	**43**	**Developer/Date: NTP 12/21/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
295018 Partial or Total Loss of CCW AK2.01 - Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER and the following: System loads				**3.3**	**3.4**
Level	**RO**	**Tier**	**1**	**Group**	**1**
General References	**ABN-20**				

Explanation	The plant was at rated power when a total loss of TBCCW occurred. Immediate Operator Actions of ABN-1 include: scram the reactor, trip 2 feedwater pumps when RPV water level begins to rise, and to verify all control rods at ≤ position 04 and power is lowering. Immediate Operator Actions of ABN-20 include: scram the reactor IAW ABN-1, and to trip all recirculation pumps. The stem states that RPV injection is 4 x 10^6 LBS/HR. Under these circumstances and those of ABN-1, 3 condensate pumps and 3 feedwater pumps are operating, with significant injection by feedwater. TBCCW cools the bearing of the condensate and feedwater pumps. With the feedwater/condensate pumps performing a lot of work with no cooling, their bearing temperatures will rise until the alarm point. ABN-20 will require further actions to prevent pump damage. Answer D is correct. If the generator were on-line with no cooling provided by TBCCW, the generator leads temperature would rise. But since ABN-1 actions have been performed, the generator is no longer on-line and no further actions are required. Answer A is incorrect. If TBCCW were lost to the recirculation MG sets, their temperatures would rise. But since ABN-20 actions have been performed, all recirculation MG sets are tripped when the recirculation pumps are tripped. There are no further actions. Answer B is incorrect. At the given conditions, the aux. cleanup pump is not in service, and it is cooled by RBCCW. Answer C is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0048 LO 274-10437	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

44 **ID: 09-1 NRO44** **Points: 1.00**

The plant was shutdown and was cooling down with the Shutdown Cooling System (SDC). Present plant conditions include the following:

- RPV water level indicates 160" and steady
- RECIRC PUMP SUCTION TEMPS indicates 340 °F
- Shutdown Cooling Pumps A and C are in service

Which of the following annunciators/indications will the **greatest** impact on the cooldown rate?

A. Annunciator 1A2 MN BRKR TRIP alarms.

B. Annunciator 1B2 MN BRKR TRIP alarms.

C. RECIRC PUMP SUCTION TEMPS rises to 351 °F.

D. SDC Pumps A and B suction pressure lowers to 10 psig.

Answer: C

Answer Explanation:

QID: 09-1 NRO44		
Question # / Answer	**44**	**Developer/Date: NTP 12/21/09**

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
295021 Loss of Shutdown Cooling AK2.03 - Knowledge of the interrelations between LOSS OF SHUTDOWN COOLING and the following: RHR/shutdown cooling					**3.6**	**3.6**
Level	**RO**	**Tier**	**1**	**Group**	**1**	
General References	**305**					

Explanation	The plant is shutdown and cooling down with SDC pumps A & B. If recirculation loop temperatures exceed 350 °F, SDC isolation valves V-17-19 & V-17-54 auto close. When V-17-19 closes, then all SDC pumps trip. This results in a total loss of SDC and the greatest impact on the cooldown rate. Answer C is correct. SDC Pump A is powered from USS 1A2, and SDC pumps B & C are powered from USS 1B2. Thus a loss of either bus results in either the A or B & C pumps available, and also impacts the cooldown rate. Answers A & B are incorrect. The SDC pumps will trip on a low suction pressure of 4 psig, not 10 as stated in answer D. Answer D is incorrect.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0045 LO 205-10445

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

45 **ID: 09-1 NRO45** **Points: 1.00**

The plant was at rated power when an event resulted in the following conditions:

- RPV water level indicates 0" and lowering slowly
- **No** RPV injection systems are available

The Steam Cooling EOP has been entered. Which of the following is correct?

IAW the EOP Users Guide, an RPV water level of ____(1)____, would still provide enough steam flow through the core to prevent exceeding ____(2)____ clad temperature.

	(1)	(2)
A.	-23"	1500 °F
B.	-28"	1500 °F
C.	-33"	1800 °F
D.	-38"	1800 °F

Answer: C

Answer Explanation:

QID: 09-1 NRO45		
Question # / Answer	**45**	**Developer/Date: NTP 12/21/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
295031 Reactor Low Water Level EK3.04 - Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL : Steam cooling	**4.0**	**4.3**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Level	RO	Tier	1	Group	1
General References	EOP Users Guide				
Explanation	The Steam Cooling EOP has been entered. Core cooling is maintained from the steam passing the uncovered portions of the fuel by one of two mechanisms: injection into the RPV is available or injection is not available. If injection is available, as long as RPV water level is $\geq$ -20", then cladding temperature will remain $\leq$ 1500 °F. If no injection is available, as long as RPV water level is $\geq$ -35", then cladding temperature will remain $\leq$ 1800 °F. With no RPV injection, an RPV water level of -33" ensures clad temperature $\leq$ 1800 °F. Answer C is correct. Answers A & B are also above -35", but the temperature limit is incorrect. Answers A & B are incorrect. Answer D is less than -35" and is incorrect.				
References to be provided during exam:	None				
Learning Objective	2621.845.0.0055 LO 3004				

Question Source (New, Modified, Bank)				Modified
Cognitive Level	Memory or Fundamental Knowledge	X 1:B	Comprehension or Analysis	
	NUREG 1021 Appendix B: Bases or purpose			
10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

46 **ID: 09-1 NRO46** **Points: 1.00**

The plant was at rated power when a manual scram was inserted due to elevated Drywell pressure. The Operator reports the following:

- Not all control rods inserted
- Reactor power indicates 55%
- MWe indicates 350

The RPV Control - With ATWS EOP directs confirming recirculation flow at minimum. IAW the EOP Users Guide, why is recirculation flow reduced instead of tripping of the Recirculation Pumps under these conditions?

A. This prevents an RPV water level shrink which could close the MSIVs.

B. This prevents an RPV water level swell which could trip the main turbine.

C. This prevents an RPV water level swell which could flood the main steam lines.

D. The Unit RO can perform this action while simultaneously inserting control rods.

Answer: B

Answer Explanation:

QID: 09-1 NRO46		
Question # / Answer	**46**	**Developer/Date: NTP 12/21/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown **EK3.01 - Knowledge of the reasons for the following responses as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Recirculation pump trip/runback**				**4.1**	**4.2**
Level	**RO**	**Tier**	**1**	**Group**	**1**
General References	**EOP Users Guide**				
Explanation	IAW the reference, the action is performed to prevent an RPV water level swell which could trip the main turbine. It is desirable to keep the turbine on-line under ATWS conditions for energy removal from the reactor because reactor power may exceed the capacity of the turbine bypass valves. All other answers are plausible but are incorrect IAW the reference. An RPV water level shrink to lo-lo would close the MSIVs, but Answer A is incorrect. Flooding the main steam lines is an undesirable condition as it may result in water hammer and steam line damage, but answer A is incorrect. While inserting control rods, the URO can reach the recirculation flow controller, but would have to stop inserting control rods to trip the recirculation Pumps. Answer D is incorrect.				
References to be provided during exam:	**None**				
Learning Objective	2621.845.0.0053 LO 3055A				

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:B**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Bases or purpose			
10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

47 **ID: 09-1 NRO47** **Points: 1.00**

The plant was shutdown with fuel shuffling in progress on the refuel floor.

The fuel hoist was loaded with a new fuel bundle in the spent fuel pool and was to be inserted into the second core quadrant. The following annunciator then alarmed while the bridge was maneuvering in the Spent Fuel Pool:

- ROD DRIFT

Which of the following states the response of the refuel bridge under the conditions provided and the basis for the refuel bridge response?

	Refuel Bridge Response	Basis
A.	The refuel bridge will be prevented from being moved anywhere over the core	Prevents a large reactivity addition into the core
B.	The refuel bridge will experiences a bridge fault and **all** refuel bridge motion will be halted	Prevents a large reactivity addition into the core
C.	The refuel bridge will be prevented from being moved anywhere over the core	Prevent damage to the fuel bundle and the drifting control rod if both were inserted at the same time in the same fuel cell
D.	The refuel bridge will experiences a bridge fault and **all** refuel bridge motion will be halted	Prevent damage to the fuel bundle and the drifting control rod if both were inserted at the same time in the same fuel cell

Answer: A

Answer Explanation:

QID: 09-1 NRO47

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Question # / Answer	47	Developer/Date: NTP 12/22/09

Knowledge and Ability Reference Information		
K&A	Importance Rating	
	RO	SRO
295023 Refueling Acc Cooling Mode AK3.02 - Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS : Interlocks associated with fuel handling equipment	3.4	3.8

Level	RO	Tier	1	Group	1

General References	UFSAR Table 7.7-1	UFSAR 9.1.4.3	
Explanation	The question describes a fuel-loaded refuel platform, currently in the fuel pool, to be moved to the second core quadrant, while a Rod Drift alarm annunciates. The annunciator shows that a control rod is no longer at position 00. When the bridge is moved toward the core area, the refuel interlock will auto stop the bridge before the core area is reached. This is to prevent a large reactivity addition to the core (largest reactivity addition would be in the cell with the control rod is drifting). Answer A is correct. As stated, bridge motion towards the core is halted, but other bridge motions are not impacted and there is no bridge fault. Answer B is incorrect. It is possible that inserting a fuel bundle into the core location with the drifting control rod, were it selected and being manually driven in by the operator (which is the expected control room action), that damage to the bundle or control rod blade could occur. None the less, it is not the basis for the refuel interlock. Answers C & D are incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.812.0.0003 LO 2391		

Question Source (New, Modified, Bank)	New

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Cognitive Level	Memory or Fundamental Knowledge	X 1: B	Comprehension or Analysis	
	NUREG 1021 Appendix B: Basis or purpose			
10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

48 **ID: 09-1 NRO48** **Points: 1.00**

The plant was at rated power with each indicated recirculation loop flow matched at 300 x 100 GPM, when the following annunciator for Recirculation Pump E alarmed:

- DRV MOT BRKR TRIP E

30 seconds later when the plant was stable, the Operator performed the IMMEDIATE OPERATOR ACTIONs IAW ABN-2, Recirculation System Failures.

Which of the following states the response of indicated RECIRC PUMP E PUMP FLOW during the following times:

1: from receipt of the annunciator until the plant stabilized 30 seconds later
2: as a result of the Operator actions

	1	2
A.	Drop to 0 GPM, then rise	Remain the same
B.	Drop to about 10 x 100 GPM	Lower to 0 GPM
C.	Drop to 0 GPM	Remain at 0 GPM
D.	Drop to 0 GPM, then rise	Lower to 0 GPM

Answer: D

Answer Explanation:

QID: 09-1 NRO48		
Question # / Answer	**48**	**Developer/Date: NTP 12/22/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

295001 Partial or Complete Loss of Forced Core Flow Circulation AA1.01 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Recirculation system	3.5	3.6

Level	RO	Tier	1	Group	1

General References	ABN-2	301.2	
Explanation	The plant was at rated power when indications show that recirculation pump E tripped. For the first 30 seconds and when the plant becomes stable, recirculation flow through the loop will lower to 0 indicated until flow through the loop goes in reverse due to the driving head of the other operating pumps. The possibility of reversed flow is noted several times in the normal procedure for the recirculation system. Both forward flow and reverse flow through a loop are treated the same and indicate identically. Thus as the tripped pump loses speed, loop flow will lower to 0. Then as reverse flow through the tripped loop is established, the tripped loop flow will rise. ABN-2 requires that the tripped pump discharge valve be closed. As the valve goes closed, the reverse flow will lower until flow has been stopped when the valve is closed. Therefore, loop E recirculation flow will lower initially, then rise somewhat, then lower as the operator action is performed to close the discharge valve. Answer D is correct. All other answers are plausible but incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.828.0.0040 LO 209		

Question Source (New, Modified, Bank)			Modified	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:PEO
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

49 **ID: 09-1 NRO49** **Points: 1.00**

The plant was at rated power with known nuclear fuel leakers and an elevated reactor coolant activity level.

A single unisolable steam/water leak occurred which resulted in **rising** indications on the TB RAGEMS, **but** the Stack RAGEMS remained **steady**.

Which of the following states the location of the leak and the type of offsite release?

	Leak Location	Release Type
A.	Condenser Bay	Ground level
B.	Reheater Protection Area	Elevated
C.	Steam Jet Air Ejector Room	Elevated
D.	Feed Pump and Condensate Pump Room	Ground level

Answer: D

Answer Explanation:

QID: 09-1 NRO49		
Question # / Answer	**49**	**Developer/Date: NTP 12/22/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
295038 High Off-site Release Rate EA1.01 - Ability to operate and/or monitor the following as they apply to HIGH OFF-SITE RELEASE RATE: Stack-gas monitoring				**3.9**	**4.2**
Level	**RO**	**Tier**	**1**	**Group**	**1**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

General References	BR 2009 sh.1, 2 BR 2011	GU 3E-661-21-1001, sh. 1	EOP Users Guide
Explanation	The plant is at power with rising trend on the TB RAGEMS but not on the stack RAGEMS. Of the locations listed, the feed/condensate pump room atmosphere discharges to the atmosphere and the reheater protection area discharge to the TB stack (and not the main stack) and both are monitored by TB RAGEMS. IAW the EOP Users Guide, a radiological release from the TB Stack (or other release point in the TB) is considered a ground level release. Answer D is correct and answer B is incorrect. The other locations listed are in the TB but discharge to the main plant stack, which is monitored by (Main) Stack RAGEMS: Condenser bay goes to the main stack (answer A is incorrect); SJAE room goes to main stack (answer C is incorrect).		
References to be provided during exam:	None		
Learning Objective	2621.828.0054 LO 288-10437		

Question Source (New, Modified, Bank)			Bank	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describe or recognize a relationship			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

50 **ID: 09-1 NRO50** **Points: 1.00**

The plant was at rated power when the Shift Manager declared that a Control Room Evacuation was required and ABN-30, Control Room Evacuation, was entered.

The Operator verified a successful reactor scram and turbine trip but off-site power failed to energize its busses. The Operator also observed the following prior to leaving the Control Room:

- EDG 1 UNIT START **and** UNIT IDLING lights are de-energized
- EDG 2 UNIT START **and** UNIT IDLING lights are de-energized

IAW 346, Operation of the Remote and Local Shutdown Panels, which of the following is required to supply power to the Station?

A. Start EDG1 at the RSP.

B. Start EDG 2 at LSP-DG2.

C. Close the S1A Startup Breaker at the RSP.

D. Close the SBO Breaker at the SBO Panel.

Answer: B

Answer Explanation:

QID: 09-1 NRO50		
Question # / Answer	50	Developer/Date: NTP 12/23/09

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
295016 Control Room Abandonment AA1.04 - Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT : A.C. electrical distribution				3.1	3.2
Level	RO	Tier	1	Group	1
General References	346	ABN-30			

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant was at rated power when a control room evacuation was required. The operator verified a successful scram and turbine trip. The stem also provided that off-site power was lost and the EDG indications provided show that neither EDG started and loaded. IAW 346, if EDG2 did not start, then manually start the EDG at the LSP-DG2. Answer B is correct. The other answers are plausible but incorrect. EDG1 cannot be started from the Remote Shutdown Panel. Answer A is incorrect. Startup breaker S1A cannot be operated from the Remote Shutdown Panel. Answer C is incorrect. Closing the SBO breaker could align the plant with the combustion turbine, it is not directed in 346.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0064 LO 308-10446

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:S**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Structures and locations			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

51 **ID: 09-1 NRO51** **Points: 1.00**

The plant was at rated power.

A leak in the service air system in the Condensate Demineralizer Regen Room, caused a drop in service air pressure and the Operator reported the following observations:

- The backup air compressors are running
- INSTR AIR SUPPLY PRESS was rising

When INSTR AIR SUPPLY PRESS reached 100 psig, the service air leak worsened. It resulted in a **lowering** of service air pressure at the rate of 2 psig/minute.

IAW ABN-35, Loss of Instrument Air, which of the following is correct under the conditions provided? (Assume plant components respond as designed)

A. **No** manual scram will be required due to lowering air pressure.

B. A manual scram will be required in 19 minutes from lowering air pressure.

C. A manual scram will be required in 23 minutes from lowering air pressure.

D. A manual scram will be required in 27 minutes from lowering air pressure.

Answer: A

Answer Explanation:

QID: 09-1 NRO51		
Question # / Answer	**51**	**Developer/Date: NTP 12/23/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

<table>
<tr><td colspan="5">295019 Partial or Total Loss of Inst. Air
AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR :
Instrument air system pressure</td><td>3.5</td><td>3.6</td></tr>
<tr><td>Level</td><td>RO</td><td>Tier</td><td>1</td><td>Group</td><td colspan="2">1</td></tr>
<tr><td>General References</td><td colspan="2">ABN-35</td><td colspan="2">RAP-M2b</td><td colspan="2">BR 2013, sh. 1</td></tr>
<tr><td>Explanation</td><td colspan="6">The plant is at power when an event caused the standby air compressors to start on low air pressure. After the compressors start, air pressure begins to recover. At 100 psig, the malfunction worsens and begins to lower air pressure at a rate of 2 psig/minute. The leak is downstream of V-6S-2, Service Air Isolation Valve. When air pressure lowers to 75 psig, the service air isolation valve, V-6S-2, will automatically close. This will isolate air going to service air from the air compressors and instrument air. Once isolated, the running air compressors will restore instrument air pressure to normal and no scram will be required due to lowering air pressure. Answer A is correct.
IAW ABN-35, when instrument air pressure drops to 55 psig, then a manual scram is required. All other answers are plausible if the candidates forget about the service isolation valve automatic action and the setpoint on which to perform the manual scram.
4/9/10: This question was modified following NRC feedback. V-6S-2 was deleted in the question stem and an air leak location downstream of V-6S-2 was inserted.</td></tr>
<tr><td colspan="2">References to be provided during exam:</td><td colspan="3">None</td><td colspan="2"></td></tr>
<tr><td>Learning Objective</td><td colspan="6">2624.828.0.0043 LO 279-10445</td></tr>
</table>

<table>
<tr><td colspan="4">Question Source (New, Modified, Bank)</td><td colspan="2">Bank</td></tr>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td colspan="2">X
3:PEO</td></tr>
<tr><td colspan="5">NUREG 1021 Appendix B: Predict an event or outcome</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td>10</td><td>55.43</td><td colspan="2"></td></tr>
<tr><td colspan="5">(SRO Only)</td></tr>
<tr><td colspan="6">Time to Complete: 1-2 minutes</td></tr>
</table>

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

52 **ID: 09-1 NRO52** **Points: 1.00**

The plant is starting up following an outage. Present plant conditions include the following:

- Reactor power is about 4%
- The turbine is being warmed through Stop Valve #2 internal bypass
- Feedwater Pump A is in service on the LFRV A

Two minutes later, RPV water level rose to 183".

Which of the following states the plant impact from this level excursion?

A. The main turbine **only** trips.

B. Feedwater Pump A **only** trips.

C. The turbine trips and the reactor scrams **only**.

D. The turbine trips, the reactor scrams, and Feedwater Pump A trips.

Answer: A

Answer Explanation:

QID: 09-1 NRO52		
Question # / Answer	**52**	**Developer/Date: NTP 12/23/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
295006 Scram AA2.03 - Ability to determine and/or interpret the following as they apply to SCRAM : Reactor water level					**4.0**	**4.2**
Level	**RO**	**Tier**	**1**	**Group**	**1**	
General References	**ABN-10 ABN-1**		**RAP-H7d RAP-H5d**		**201 317**	

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant is starting up with power at about 4%, and the main turbine is being heated. In order to preheat the main turbine through stop valve #2, the turbine (trip) must be reset. Once reset, the turbine will trip from a turbine trip signal. As RPV water rises to 175", a turbine trip signal is generated. As water level continues to rise to 181", Reactor Overfill Protection System (ROPS) can be activated, unless bypassed. When activated, all operating feedwater pumps trip. ROPS is automatically bypassed when total feedwater flow is < 2.23×10^6 lb/hr. IAW procedures 201 and 317, feedwater flow should be transferred to the main flow regulation valves (MFRV) at about 0.6×10^6 lb/hr. Therefore, with only 1 feedwater pump operating at < 0.6×10^6 lb/hr, then ROPS is automatically bypassed on low feedwater flow. As the ROPS high RPV water level setpoint is reached, the operating feedwater pump will not trip since ROPS is bypassed. Answer A is correct. Answer B is incorrect since the feedwater pump does not trip. As discussed, the turbine will trip but the reactor will not scram on the high water level nor will it trip from the turbine trip (bypassed at <40%). Answer C & D are incorrect. The reactor will scram on RPV low water level of 138" There is no direct scram from RPV high water level.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0051 LO 249-10445	

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

53 **ID: 09-1 NRO53** **Points: 1.00**

The plant was at rated power. A timeline of events is shown below (hh:mm):

0800 Breaker 1A **inadvertently** opened
0815 Annunciator MN BRKR 1C 86 LKOUT TRIP alarmed
0830 The reactor scrammed on high Drywell pressure

Which of the following is correct for the given conditions?

A. Feedwater Pump A is **not** available for RPV injection.

B. Core Spray loops A **and** B are available for RPV injection.

C. **Both** Service Water Pumps are available to provide cooling water.

D. Containment Spray Pumps 51C **and** 51D are available for Drywell Sprays.

Answer: D

Answer Explanation:

QID: 09-1 NRO53		
Question # / Answer	53	Developer/Date: NTP 12/23/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
295003 Partial or Complete Loss of AC AA2.04 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : System lineups					3.5	3.7
Level	RO	Tier	1	Group	1	
General References	RAP-S1e		BR 3002, sh. 2		RAP-T2a	

Explanation	The plant was at rated power when breaker 1A inadvertently opened. When this breaker opens, Startup Breaker S1A automatically closes and all busses remain energized. At 0815, a lockout on Bus 1C causes Bus 1C occurs and Bus 1C will de-energize and the downstream busses (USS 1A1, USS 1A2, and USS 1A3) will also de-energize. Because there is a fault on the 1C bus, EDG1 will not fast start and load onto the bus, but remains off. When the plant scrams, Bus 1B power is automatically transferred to Startup breaker S1B. So over the events, Bus 1C is de-energized (and the downstream busses). Containment Spray Pumps are available (powered from USS 1B2) and the associated ESW Pumps are also available (powered from Bus 1D). Answer D is correct. As stated, when breaker 1A opens, the S1A breaker closes and Bus 1A remains energized (with no impact from the scram). Thus Feedwater Pump A (powered from Bus 1A) is available. Answer A is incorrect. Core Spray Main Pump A (powered from Bus 1C) and Booster Pump A (powered from USS 1A2) are not available. The Core Spray B loop does have power. Answer B is incorrect. Service Water Pump 1-1 (powered from USS 1A3) is not available, whereas Service Water Pump 1-2 (powered from USS 1B3) is available, but not both. Answer C is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0009 LO 226-10453	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:DR**
	NUREG 1021 Appendix B: Describing or recognizing relationships			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

54 **ID: 09-1 NRO54** **Points: 1.00**

The plant was at rated power when an event occurred. Present plant conditions include the following:

- An ATWS is in progress
- RPV pressure is 1000 psig and stable
- EMRV NR108A is stuck full open
- 3 Turbine Bypass Valves indicate full open
- APRM indication has been lost
- The Isolation Condensers are in Standby
- Torus water temperature is 100 °F and is rising at 2 °F/minute
- Torus Cooling is inoperable

IAW the EOP Users Guide, which of the following states the **maximum** time until SLC injection is required (Assume that reactor power remains unchanged over the times listed)?

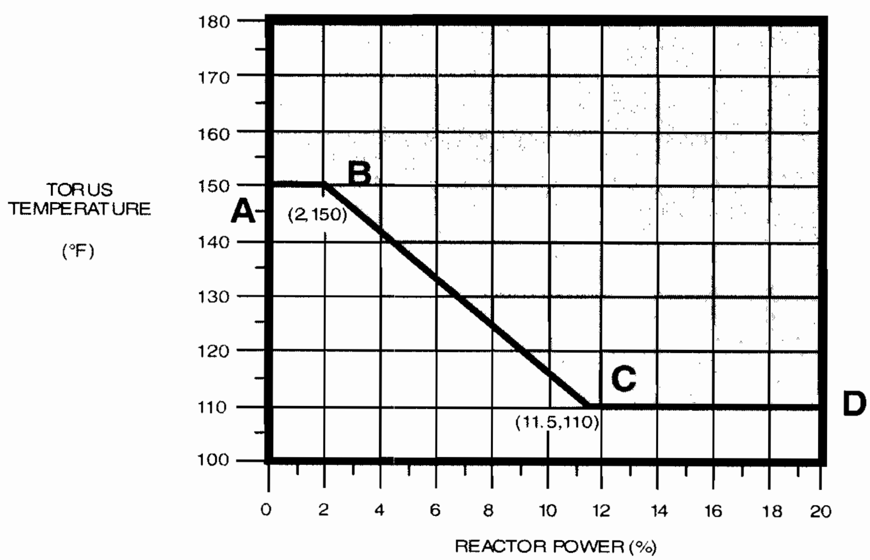


FIG L BIIT
BORON INJECTION
INITIATION TEMPERATURE

A. 5 minutes

B. 6 minutes

C. 7 minutes

D. 8 minutes

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer: A

Answer Explanation:

QID: 09-1 NRO54		
Question # / Answer	**54**	**Developer/Date: NTP 12/23/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
295026 Suppression Pool High Water Temp 2.4.47 - Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.					**4.2**	**4.2**
Level	**RO**	**Tier**	**1**	**Group**	**1**	
General References	**EOP Users Guide**	**UFSAR Table 5.1-1**		**UFSAR 7.7.1.5**		
Explanation	The plant was at power when an event occurred: an ATWS with a rising torus water temperature. APRM indication has been lost. But power can be estimated enough to determine when SLC injection is required. The capacity of the turbine bypass valves is 40%, or 5% per bypass valve. With 3 turbine bypass valves open, and RPV pressure stable at 1000 psig, then reactor power is about 15%. But a full open EMRV at 1250 psig allows over 600,000 lb/hr, and a reduced flow rate at 1000 psig. Therefore, the total steam flow would equate to > 15% reactor power. SLC injection is required when the Torus temperature cannot be maintained below the BIIT temperature for the given reactor power. From the BIIT Curve provided, at any power > 12%, the Curve will be met when torus water temperature reaches 110 ^oF. Thus, after 5 minutes, torus temperature will be 110 ^oF at a power > 12% and the BIIT Curve will be met, requiring SLC injection. Answer A is correct. The other answers are plausible if reactor power is estimated incorrectly or graph misread.					
References to be provided during exam:	**None**					

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Learning Objective	2621.845.0.0053 LO 3055A

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**x 3:SPR**
	NUREG 1021 Appendix B: Solve a problem using references			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

55 **ID: 09-1 NRO55** **Points: 1.00**

Consider the events listed below with the reactor at power. The reactor **remained** at power for each event.

Which of the following events represents a violation of Tech Specs?

A. RPV water level of 139" at 25% reactor power.

B. Drywell pressure of 3.2 psig at 15% reactor power.

C. Turbine Stop Valve closure at 45% reactor power.

D. RPV pressure of 1054 psig at 55% reactor power.

Answer: C

Answer Explanation:

QID: 09-1 NRO55		
Question # / Answer	55	Developer/Date: NTP 12/23/09

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
295005 Main Turbine Generator Trip 2.2.42 - Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.				3.9	4.6
Level	RO	Tier	1	Group	1
General References	TS Table 3.1.1	TS 2.3			

Explanation	A turbine stop valve closure (which occurs during a turbine trip) scram is not required below 40% rated thermal power. Thus, if the reactor does not scram at 45% power, this would result in a TS violation and LCO entry. Answer C is correct. TS requires a scram when RPV water level is > 11'5" (137"). With RPV water level at 139" while at power, no Tech Spec has been violated. Answer A is incorrect. TS requires a scram from Drywell pressure of ≤ 3.5 psig. Thus, no scram with Drywell pressure of 3.2 psig does not violate TS. Answer B is incorrect. TS requires a scram at ≤ 1060 psig. With RPV pressure at 1054 psig while at power, no Tech Spec has been violated. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.850.0.0090 LO 1658	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:D**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Definitions			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

56 **ID: 09-1 NRO56** **Points: 1.00**

The plant was at rated power when the Control Room was notified that Drywell pressure switches PS RV46A and PS RV46B, which inputs into the starting circuit for the Core Spray System, have failed in its current state such that they will not detect a high Drywell pressure condition.

Which of the following states the ability of the Core Spray System to function during a high Drywell pressure condition?

A. Core Spray Pumps A **and** B will auto start as designed, with no manual Operator actions required.

B. Core Spray Pump A will **not** auto start, but **may** be manually started. Core Spray Pump B will auto start as designed.

C. Core Spray Pump A will **not** start and **cannot** be manually started. Core Spray Pumps B **and** C auto start as designed.

D. **Neither** Core Spray Pump A **or** B will auto start, but can be manually started. All other Core Spray components operate as designed.

Answer: A

Answer Explanation:

QID: 09-1 NRO56		
Question # / Answer	**56**	**Developer/Date: NTP 12/23/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
295024 High Drywell Pressure 2.2.37 - Ability to determine operability and/or availability of safety related equipment.						**3.6**	**4.6**
Level	**RO**	**Tier**	**1**	**Group**	**1**		
General References	**NU 5060E6003, sh. 1-4**	**2621.828.0.0010**		**RAP-C2f**			

Explanation	With no failures, a single high Drywell pressure signal will start the Core Spray System normally. This includes the Core Spray System A and B. There are 4 Drywell high pressure switches. If any two fails, there are still 2 others to start the Core Spray System in its normal start mode. Answer A is correct. Two instrument failures in RPS could render that RPS channel inoperable, but the Core Spray start logic is inter-mixed among systems. The other answers are plausible but incorrect since no manual operator actions are required.
References to be provided during exam:	None
Learning Objective	2621.828.0.0010 LO 209-10439

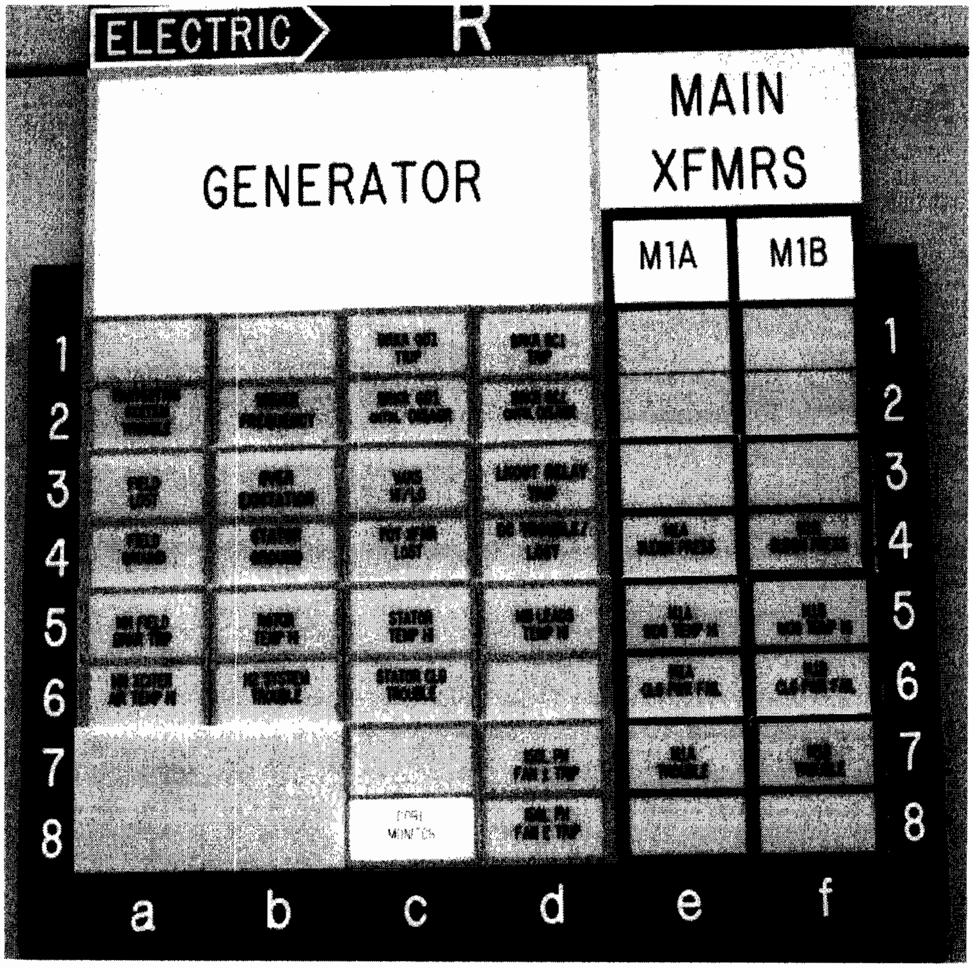
Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

57 **ID: 09-1 NRO57** **Points: 1.00**

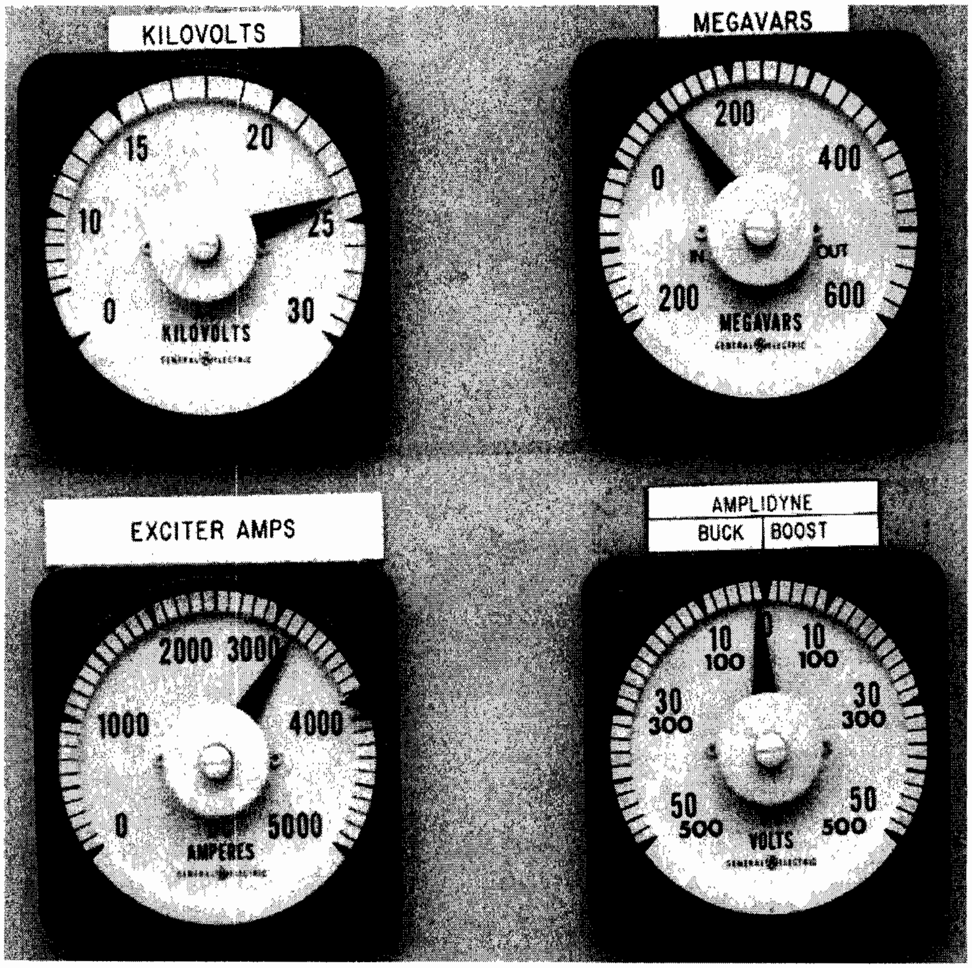
The plant is at rated power, and the electrical plant is in a normal lineup.

Which of the following indications or alarms, by themselves, would require entry into and performance of ABN-12, Generator Excitation Equipment Malfunction?

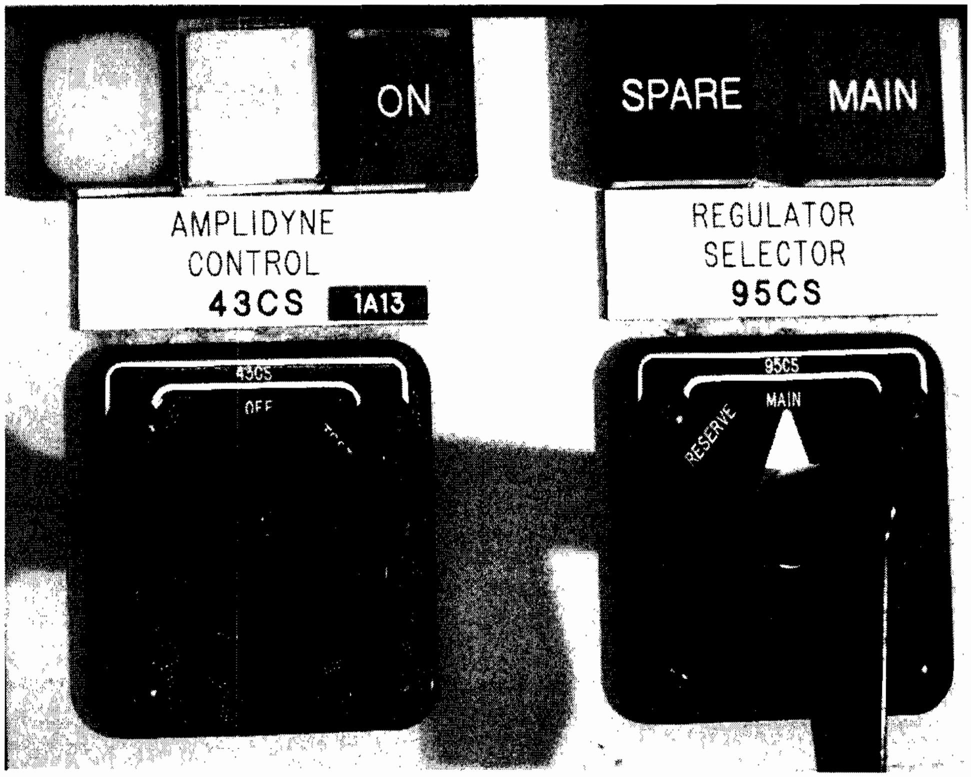
A.



B.

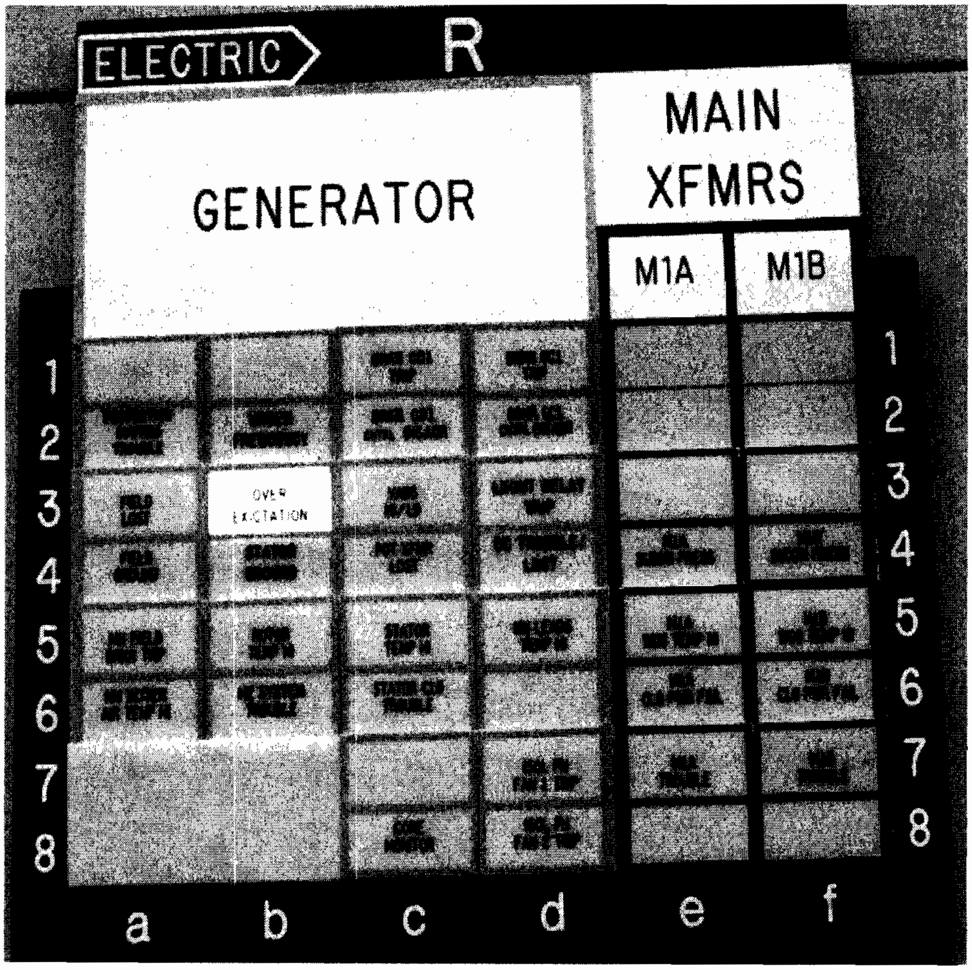

KILOVOLTS
15
20
10
25
0
30
KILOVOLTS
MEGAVARS
200
400
0
IN
OUT
200
600
MEGAVARS
EXCITER AMPS
2000
3000
1000
4000
0
5000
AMPERES
AMPLIDYNE
BUCK
BOOST
10
100
0
10
100
30
300
30
300
50
500
50
500
VOLTS

C.



ON
SPARE
MAIN
AMPLIDYNE
CONTROL
43CS
1A13
REGULATOR
SELECTOR
95CS
43CS
OFF
95CS
MAIN
RESERVE

D.



Answer: C

Answer Explanation:

QID: 09-1 NRO57		
Question # / Answer	**57**	**Developer/Date: NTP 12/26/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

700000 Generator Voltage and Electric Grid Disturbances **2.4.4 - Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures.**						**4.5**	**4.7**
Level	**RO**	**Tier**	**1**	**Group**	**1**		
General References	**ABN-12**		**336.1**		**RAP-R3b**		
Explanation	ABN-12 applies to the following events: 1) trip of the generator voltage regulator (amplidyne); 2) erratic operation of generator voltage regulation equipment (amplidyne); 3) Loss of 125 VDC control power to excitation switchgear. Answer C shows the amplidyne control switch in the position for automatic control but the red light is off and the green light is on which indicates the amplidyne has tripped. Answer C is correct. Answer A shows that the core monitor is in alarm and can be an indication of over-heating or insulation breakdown in the generator. This can be caused by load irregularities related to the amplidyne. ABN-12 does list this annunciator as an indication once the ABN is entered. But by itself, it is not an entry. Answer A is incorrect. Answer B shows normal full load indications for volts and vars. If the amplidyne were faulty, it could result in generator voltage out of normal limits (23.3 - 24.7 KV). Answer B is incorrect. Answer D could be a result of a faulty amplidyne, and it is noted in ABN012 as an indication. But by itself, it is not an entry into ABN-12. Answer D is incorrect.						
References to be provided during exam:	**None**						
Learning Objective	2621.828.0.0025 LO 248-10445						

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X** **3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			

10CRF55 Content	55.41	10	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

58 **ID: 09-1 NRO58** **Points: 1.00**

The plant was at rated power, when a large LOCA inside the Primary Containment occurred.

IAW the EOP Users Guide, elevated temperatures in the Drywell can lead to elevated temperatures in the RPV water level instruments' ________(1)______ leg which may result in a false indicated RPV water level. This false indicated RPV water level could ________(2)______ .

	(1)	(2)
A.	reference	result in Core Spray initiation
B.	variable	require manual tripping of CRD Pumps
C.	reference	result in ROPS initiation
D.	variable	result in Isolation Condenser initiation

Answer: C

Answer Explanation:

QID: 09-1 NRO58		
Question # / Answer	**58**	**Developer/Date: NTP 12/26/09**

Knowledge and Ability Reference Information					
K&A				**Importance Rating**	
				RO	**SRO**
295028 High Drywell Temperature EK2.02 - Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Components internal to the drywell				**3.2**	**3.3**
Level	**RO**	**Tier**	**1**	**Group**	**1**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

General References	**EOP Users Guide**	**RAP-H5d**	
Explanation	The plant was at rated power when a large LOCA occurred inside the Drywell. Elevated Drywell temperatures can lead to elevated temperatures in the RPV water level instrument reference legs and a higher than normal indicated water level due to the change in water density. ROPS initiation occurs on an RPV high water level signal (181"). Answer C is correct. Since the variable leg is incorrect, then answers B & D are incorrect. Also, ABN-1 requires tripping CRD Pumps if RPV water level cannot be maintained below 170". Answer A has the correct level leg but Core Spray starts on an RPV low water level signal, not high. Answer A is incorrect.		
References to be provided during exam:	**None**		
Learning Objective	2624.845.0.0052 LO 3053		

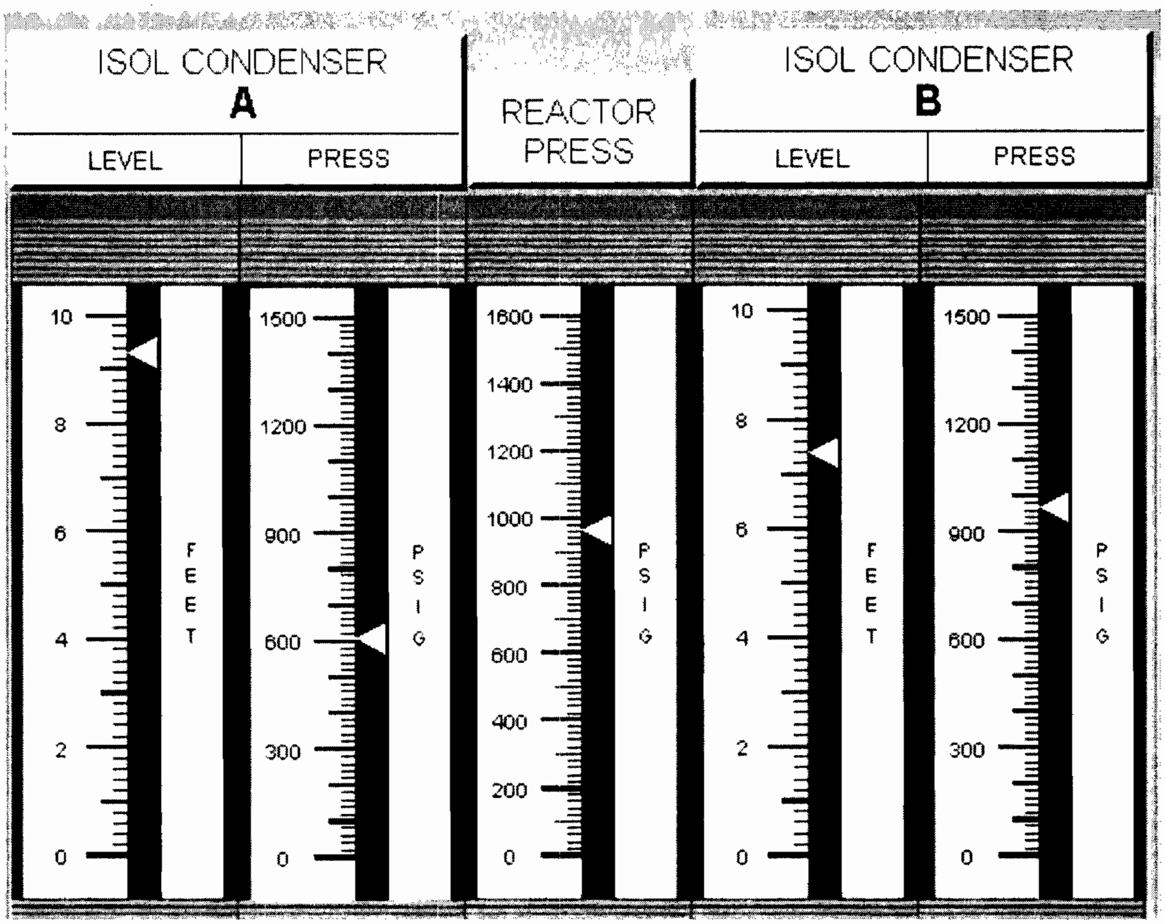
Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:RI**
	NUREG 1021 Appendix B: Recognize interaction between systems, including consequences and implications			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

59 **ID: 09-1 NRO59** **Points: 1.00**

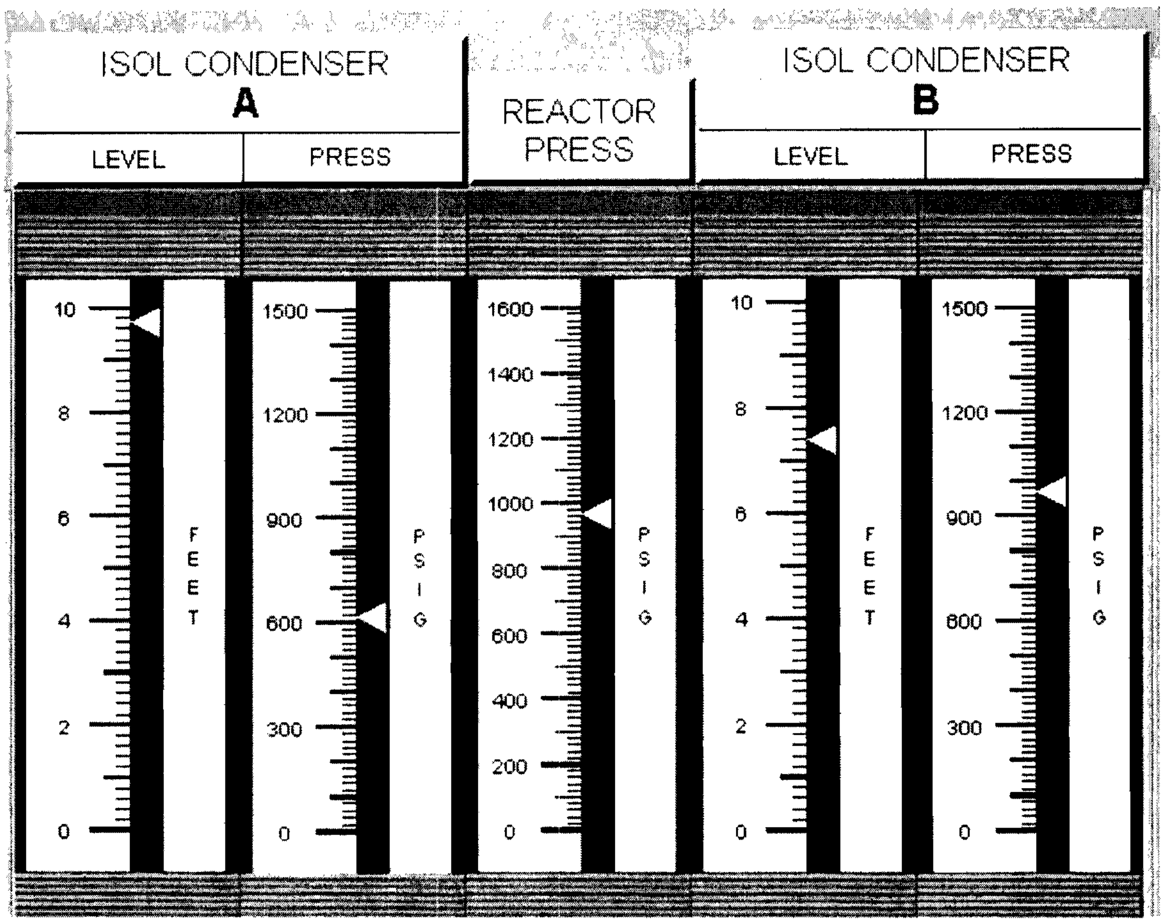
The plant was at rated power with elevated offgas radiation readings, when an event occurred.

The Operator reported the following alarms and indications:

- Annunciator AREA MON HI has alarmed and the ISOLATION COND AREA ARM indicated 80 mr/hr and steady
- Annunciator COND A FLOW HI POSSIBLE RUPTURE has alarmed
- See the indications below



Five minutes later, the Operator observes the following indications:



Which of the following is correct?

A. A radiological release **was** in progress and Isolation Condenser A **had** successfully isolated.

B. A radiological release **is** in progress and Isolation Condenser A **has** failed to automatically isolate and should be manually isolated.

C. An Isolation Condenser A steam leak into the reactor Building occurred and Isolation Condenser A **had** successfully isolated.

D. An Isolation Condenser A steam leak into the Reactor Building occurred and Isolation Condenser A **has** failed to automatically isolate.

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer: B

Answer Explanation:

QID: 09-1 NRO59		
Question # / Answer	59	Developer/Date: NTP 12/26/09

Knowledge and Ability Reference Information							
K&A						Importance Rating	
						RO	SRO
295017 High Off-site Release Rate AK1.02 - Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE : Protection of the general public						3.8	4.3
Level	RO	Tier	1	Group	2		
General References	RAP-C3a						

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

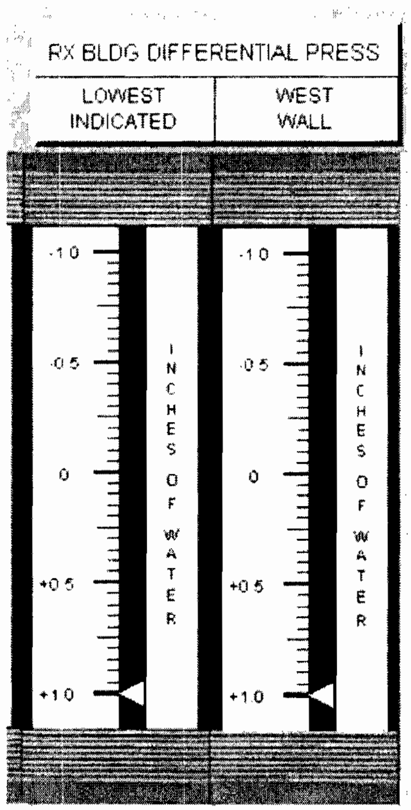
Explanation	The plant was at power when indications are provided which show the following: Isolation Condenser A shell water level is high and pressure is low. The Isolation Condenser area radiation monitor is alarming, and the Possible Rupture alarm is in alarm. Two minutes later, indications are shown again. This indication shows that Isolation Condenser A shell water level is even higher. When the Possible Rupture comes in, after a 27 second time delay, the Isolation Condenser A isolation valves will go closed. Thus, about 1.5 minutes after this alarm, the valves must still be open since the shell water level is still rising. Therefore, there is a tube leak on Isolation Condenser A resulting in an offsite release (the shell directly vents outside of the reactor Building), the isolation has failed, and the Operator should manually perform the actions to isolate the Isolation Condenser. Answer B is correct. If the candidate does not realize that shell water level is still rising, then it will appear that the condenser automatically isolated. Answer A is incorrect. The possible Rupture annunciator could also be indicative of a steam line break in the Reactor Building, especially with the Isolation Condenser ARM in alarm. But since a steam line break would not impact shell water level as in the question, answers C & D are incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0023 LO 2338	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

60 **ID: 09-1 NRO60** **Points: 1.00**

The plant was at rated power when an unisolable steam leak began in the Reactor Building. The Operator reports the following observations (see below):



Which of the following states the status of Reactor Building HVAC and the Standby Gas Treatment System (SGTS)? (Assume **no** Operator actions)

	RB HVAC	SGTS
A.	Tripped	In Standby
B.	Tripped	Running
C.	Running	In Standby
D.	Running	Running

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Answer: A

Answer Explanation:

QID: 09-1 NRO60		
Question # / Answer	60	Developer/Date: NTP 12/28/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
295035 Secondary Containment High Differential Pressure EK2.01 - Knowledge of the interrelations between SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE and the following: Secondary containment ventilation					3.6	3.6
Level	RO	Tier	1	Group	2	
General References	GE 157B6350 sh. 72A	BR 3017		2621.828.0.0042		
Explanation	The plant is at power when an unisolable leak starts in the RB. The Operator reports that RB Dp is +1.0 inches/water. At this level, the normal RB HVAC trips to prevent over-pressurizing the RB. The same signal has no input into the auto start of SGTS and it remains in standby. Answer A is correct. All other answers are plausible but incorrect. Note that RB HVAC and SGTS can run simultaneously.					
References to be provided during exam:	None					
Learning Objective	2621.828.0.0042 LO 261-10445					

Question Source (New, Modified, Bank)				Modified
Cognitive Level	Memory or Fundamental Knowledge	X 1:I	Comprehension or Analysis	
	NUREG 1021 Appendix B: Interlocks, setpoints or system response			
10CRF55 Content	55.41	7	55.43	
	(SRO Only)			

Time to Complete: 1-2 minutes

61 **ID: 09-1 NRO61** **Points: 1.00**

The plant was at rated power when an unisolable steam leak began in the Reactor Building, and the Secondary Containment Control EOP was entered.

Conditions worsened and the SRO has directed Emergency Depressurization due to the radiation levels in the Reactor Building.

IAW the EOP Users Guide, which of the following states the bases for directing an Emergency Depressurization?

1. It places the RPV in the lowest energy state
2. It reduces the driving head on the leak
3. It allows low pressure systems to inject into the RPV
4. It reduces the amount of energy available to be deposited inside the Primary Containment

A. 1 and 3

B. 1 and 2

C. 2 and 4

D. 1, 2, and 4

Answer: B

Answer Explanation:

QID: 09-1 NRO61		
Question # / Answer	**61**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
295033 High Secondary Containment Area Radiation Levels EK3.01 - Knowledge of the reasons for the following responses as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Emergency depressurization	**3.3**	**3.5**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Level	RO	Tier	1	Group	2

General References	**EOP Users Guide**		
Explanation	IAW the EOP Users Guide, the radiation increases is so wide spread that is poses a direct threat to secondary containment integrity, equipment located in the secondary containment or continued safe operation. ED will place the plant in its lowest energy state and will reduce the driving head and flow from primary systems that are discharging into the secondary containment. Answer B (1 and 2) is correct. Answer A is incorrect since it does not state both reasons for ED and lists a reason not associated with the basis for ED. ED is performed by opening the EMRVs which releases the energy from the RPV into the Torus. ED does not reduce the amount of energy to be released to the primary containment but into the Secondary Containment. Thus selection 4 is incorrect and answer D is incorrect. It is true that as RPV pressure lowers from the ED, more injection systems become available to inject into the RPV to compensate for the leak. But, this is not a reason for the ED in the question. Selection 3 is incorrect and answer C is incorrect. **Note**: A question on the most previous ILT NRC exam asked why ED is required due to secondary containment temperatures (and this question refers to radiation levels, and the question is basically the same question) 4/9/10: This question was modified due to NRC feedback.		
References to be provided during exam:	**None**		
Learning Objective	2621.845.0.0057 LO 3082		

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:B**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Bases or purpose			

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

10CRF55 Content	**55.41**	**5**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

62 **ID: 09-1 NRO62** **Points: 1.00**

Refueling was in progress when an accident occurred on the refuel floor. The Standby Gas Treatment System initiated **immediately** when sensed radiation levels went above the setpoint.

Which of the following states the logic necessary for the SGTS automatic start described above?

A. Radiation levels $\geq$ 9 mr/hr as sensed by **both** RB Vent Radiation Monitors.

B. Radiation levels $\geq$ 9 mr/hr as sensed by **either** RB Vent Radiation Monitor.

C. Radiation levels $\geq$ 50 mr/hr as sensed by **both** Refuel Floor Radiation Monitors.

D. Radiation levels $\geq$ 50 mr/hr as sensed by **either** Refuel Floor Radiation Monitors.

Answer: B

Answer Explanation:

QID: 09-1 NRO62		
Question # / Answer	**62**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
295034 Secondary Containment Ventilation High Radiation **EA1.04 - Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : SBGT/FRVS**				4.1	4.2
Level	**RO**	**Tier**	**1**	**Group**	**2**
General References	**651.4.001**				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	SGT will auto start from ≥ 9 Mr/hr on EITHER RB vent radiation monitor (with no time delay), OR high radiation (50 Mr/hr) on the refuel floor rad monitors (with a 2-minute time delay). Since there was an immediate SGTS start, the RB vent rad monitors must have sensed rad levels above their setpoint. Therefore, only one RB vent radiation monitor had to reach its setpoint of 9 mr/hr to automatically auto start SGTS. Answer B is correct. The other answers are plausible if the SGTS start logic is not understood.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0042 LO 261-10450	

Question Source (New, Modified, Bank)			**Bank**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:I**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Interlocks, setpoints, or system response			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

63 **ID: 09-1 NRO63** **Points: 1.00**

The plant was starting up after an outage. The following conditions currently exist:

- All IRMs are mid-range on Range 8 and **stable**
- Turbine warming is in-progress

The Operator reports the following:

- Annunciator COND VAC LO 25 INCHES has alarmed
- CONDENSER VACUUM 1A, 1B and 1C indicated 24" HG and are **degrading** at a rate of ½ "HG/minute
- RPV pressure is 580 psig and **stable**

Which of the following is correct given the above conditions (assume **no** operator actions)?

A. The reactor will **not** scram on low condenser vacuum.

B. The turbine trip will generate a reactor scram in 4 minutes.

C. A low condenser vacuum will generate a reactor scram in 4 minutes.

D. An RPV high pressure will generate a reactor scram in 8 minutes when the Turbine Bypass Valves close.

Answer: A

Answer Explanation:

QID: 09-1 NRO63		
Question # / Answer	**63**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
295002 Loss of Main Condenser Vac AA2.02 - Ability to determine and/or interpret the following as they apply to LOSS OF MAIN CONDENSER VACUUM : Reactor power	**3.2**	**3.3**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Level	RO	Tier	1	Group	2
General References	RAP-G4c		RAP-J1b		

Explanation	The question stem shows that the reactor is starting up (with the mode switch in STARTUP) with RPV pressure at 580 psig and steady. At this low pressure, the main condenser low vacuum scram signal **and** the turbine stop valve closure scram signal are bypassed. The low vacuum scram setpoint is 22" hg, and the turbine bypass valves close at 10" hg. In 4 minutes, condenser vacuum drops to 22" hg, which is the scram and turbine trip setpoint (currently bypassed). At this low RPV pressure, the low vacuum signal is bypassed and the reactor will not scram from a low vacuum scram signal or from the turbine trip, and power remains at its current power level. Answer A is correct. The turbine will receive a trip signal in 4 minutes (when condenser vacuum lowers to 22") but the scram is bypassed since RPV pressure is < 600 psig. Answer B is incorrect. Condenser vacuum will lower to 22" in 4 minutes, but this scram signal is also bypassed. Answer c is incorrect. When condenser vacuum reaches 10" (or in 28 minutes later), the turbine bypass valves will close which could result in a rising RPV pressure to the scram setpoint. But, in 8 minutes only, the low condenser vacuum trip #2 to close the turbine bypass valves has not yet been reached. Answer D is incorrect.
References to be provided during exam:	None
Learning Objective	2621.828.0.0037 LO 212-10445

Question Source (New, Modified, Bank)				**Modified**
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:DR**
	NUREG 1021 Appendix B: Describe or recognize relationships			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

64 **ID: 09-1 NRO64** **Points: 1.00**

The plant was at 60% power during power ascension following a 30-day refuel outage. The A Recirculation Pump was out of service and was in an idle configuration.

An event then occurred which resulted in a rising RPV pressure. The Operator reported the following:

- RPV pressure resulted in several EMRVs opening for 5 seconds. All AUTO DEPRESS VALVE indications currently indicate **green** light on
- The reactor scrammed and **all** IMMEDIATE OPERATOR ACTIONs of ABN-1, Reactor Scram, were completed

With **no** further Operator action, which of the following is correct?

A. RPV water level is controlling at 142" with the MFRV in AUTO.

B. RPV pressure will **lower** due to the initiation of the Isolation Condensers.

C. RPV pressure will be maintained by the Electronic Pressure Regulator (EPR).

D. **All** reactor Recirculation Pumps tripped **immediately** upon reaching the high pressure setpoint.

Answer: B

Answer Explanation:

QID: 09-1 NRO64		
Question # / Answer	**64**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**
295007 High Reactor Pressure **2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.**	**3.94.4**	**4.7**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Level	RO	Tier	1	Group	2

General References	ABN-1	RAP-C1a RAP-E1a	RAP-B4g
Explanation	The plant was at 60% power following a refuel outage when an event occurred which resulted in several EMRVs opening for 5 seconds, then closing. The lowest pressure setpoint to open EMRVs is 1065 psig. The isolation condensers auto initiate at an RPV pressure of 1051 psig. Thus the Isolation Condensers have initiated. with the isolation condensers in service and a small decay heat load, the RPV will depressurize. Answer B is correct. IAW ABN-1, when RPV water level begins to rise, it directs selecting 1 Feedwater Pump and tripping the others. It then directs placing all MFRVs in manual and taking them to close (all LFRVs would be closed at this power level). This will terminate Feedwater injection into the RPV. Thus, RPV water level will not be controlled with the MFRV in AUTO. Thus answer A is incorrect. Sine the reactor was starting up after a refuel outage, there is little decay heat to maintain RPV pressure following the scram and RPV pressure is lowering uncontrollably due to the Isolation Condensers, RPV pressure is not being maintained by the EPR. Answer C is incorrect. The reactor recirculation pumps trip on RPV high pressure of 1051 psig. At this pressure, pumps A, B & E trip immediately, and pumps C & D trip if the high pressure is sustained for 10.5 seconds. Therefore, all pumps may have tripped but not all immediately. Answer D is incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.828.0.0023 LO 2338		

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describe or recognize relationships			

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

10CRF55 Content	55.41	5	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

65 **ID: 09-1 NRO65** **Points: 1.00**

The plant was at 775 psig during a startup. Power ascension was halted to determine the cause of the over-current trip of CRD Pump B.

15 minutes later, the following annunciators then alarmed:

- 1A2 MN BRKR TRIP
- 1A2 MN BRKR OL TRIP

Which of the following actions is required?

A. Place RPS MG Set 1 on Transformer PS-1.

B. Secure Reactor Building normal HVAC and start Standby Gas Treatment System I.

C. Depress both Manual Scram pushbuttons and place the Reactor Mode Selector switch in SHUTDOWN immediately.

D. Depress both Manual Scram pushbuttons and place the Reactor Mode Selector switch in SHUTDOWN when 2 HCU accumulator alarms are received.

Answer: C

Answer Explanation:

QID: 09-1 NRO65		
Question # / Answer	**65**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
295022 Loss of CRD Pumps 2.2.2 - Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.					**4.6**	**4.1**
Level	**RO**	**Tier**	**1**	**Group**	**2**	
General References	**RAP-H1c**		**ABN-45**			

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Explanation	The plant is at 775 psig during a startup, which has been suspended to diagnose the trip of CRD Pump B. With this pump OOS, CRD Pump A will be started. Then, a lockout occurs on USS 1A2, which powers CRD Pump A. Thus, both CRD Pumps are lost and will not be immediately restored. IAW RAP-H1c, if RPV pressure is <850 psig, and charging pressure cannot be immediately re-established, then scram IAW ABMN-1, reactor Scram. Answer C is correct. Placing RPS MG Set 1 loads on Transformer PS-1 is required IAW ABN-45, but the MG itself is not re-started on the transformer. Answer A is incorrect. ABN-45 directs shutdown on RB normal ventilation and manual start of SGTS II - not System I. Answer B is incorrect. IAW RAP-H1c, with RPV pressure >850 psig and no CRD Pumps available, if charging pressure cannot be immediately re-established, and 2 or more accumulator trouble alarms are received, then a manual scram IAW ABN-1 is required. Answer D is incorrect.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0011 LO 10465

Question Source (New, Modified, Bank)				**Modified**
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:DR**
	NUREG 1021 Appendix B: Describe or recognize relationships			
10CRF55 Content	**55.41**	**6**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

66 **ID: 09-1 NRO66** **Points: 1.00**

The plant is at rated power.

The URO, BOP, US, SM and an inactive licensed RO from the tagging desk were in the Control Room. The URO will be leaving the Control Room to eat lunch in the cafeteria.

IAW OP-OC-101-111-1001, Strategies for Successful Transient Mitigation, which of the following is correct?

A. A turnover from the URO to the BOP is **not** required.

B. The URO shall turnover to the BOP, **but not** to the tagging RO.

C. The URO **cannot** leave the Control Room since minimum shift staffing will not be met.

D. The URO turnover shall be documented in the Control Room Log.

Answer: B

Answer Explanation:

QID: 09-1 NRO66		
Question # / Answer	**66**	**Developer/Date: NTP 12/28/09**

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Conduct of Operations **2.1.18 Ability to make accurate, clear and concise logs, records, status boards, and reports.**					3.6	3.8
Level	RO	Tier	3	Group		
General References	OP-OC-101-111-1001	OP-OC-100-1001				

Explanation	IAW the reference, the URO shall turnover his responsibilities to the BOP if the URO will leave the 'at the controls area' for an extended time. Although another licensed RO is in the control room at the time, this RO is not an active license holder and cannot stand watch alone. Therefore, B is correct and A is incorrect. IAW OP-OC-100-1001, only 1 RO is required for minimum shift staffing. Answer C is incorrect. There is no requirement to log the URO to BOP turnover IAW OP-OC-101-111-1001. Logging shift turnover at the start of each shift is required. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

67 **ID: 09-1 NRO67** **Points: 1.00**

The plant was at rated power when an event occurred which required the Operator to perform a rapid power reduction to 80% power. This action stabilized the plant.

IAW 202.1, Power Operation, which of the following shall be notified of the power reduction, in order to perform a Tech Spec required action?

A. Licensing

B. Chemistry

C. Rad Protection

D. Reactor Engineering

Answer: B

Answer Explanation:

QID: 09-1 NRO67		
Question # / Answer	**67**	**Developer/Date: NTP 12/29/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
Conduct of Operations 2.1.38 Knowledge of the station's requirements for verbal communications when implementing procedures.					**3.7**	**3.8**
Level	**RO**	**Tier**	**3**	**Group**		
General References	**202.1**	**TS 3.6.A.4**				
Explanation	The plant was at rated power when a 20% power reduction was performed by a rapid power reduction. IAW 201 and TS, if reactor power changes by $\geq$15% (289.5 MWth) in an hour, then Chemistry must be notified in order to sample and analyze reactor coolant. Answer B is correct. The other answers are plausible but incorrect.					

EXAMINATION ANSWER KEY

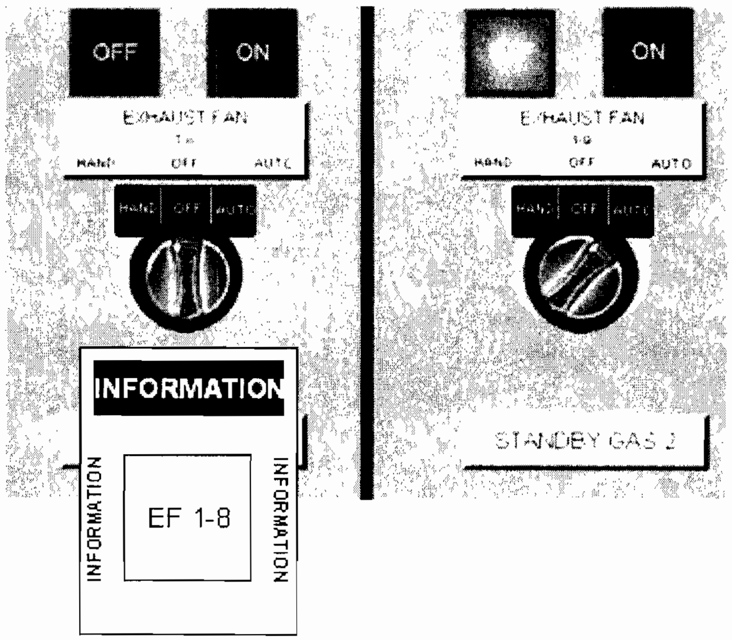
ILT 09-1 NRC RO Exam

References to be provided during exam:	**None**	
Learning Objective		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

68 **ID: 09-1 NRO68** **Points: 1.00**

The plant was at rated power. You have just come in for your second day shift. On your Control Room tour during turnover, you note the change in status in the Standby Gas Treatment System (SGTS) as shown below:



Which of the following maintenance activities, if it resulted in tripping the breaker, will impact the LCO for the SGT System?

Troubleshooting on the feeder breaker to ...

A. USS 1A2

B. USS 1B2

C. USS 1B3

D. VMCC 1B2

Answer: B

Answer Explanation:

QID: 09-1 NRO68

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Question # / Answer	68	Developer/Date: NTP 12/29/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Equipment Control **G2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.**					3.1	4.2
Level	RO	Tier	3	Group		
General References	TS 3.5.B.7		330		BR 3002 sh. 2	
Explanation	The plant is at rated power with indications showing that SGTS Fan 1 is inoperable with its breaker open and tagged out of service. TS 3.5.B.6 allows a 7-day LCO with one SGTS train inoperable. If both trains were inoperable, then TS 3.5.B.7 will require a 24-hour shutdown. SGTS fan B (EF 1-9) is powered from MCC 1B24, which is fed from USS 1B2. Thus, troubleshooting on the feeder breaker to USS 1B2 has the potential to de-energize USS 1B2, which would inop the only remaining SGTS fan, and TS 3.5.B.7 would apply for SGTS. Answer B is correct. All other answers are plausible but incorrect.					
References to be provided during exam:	None					
Learning Objective	2624.828.0.0042 LO 261-10445					

Question Source (New, Modified, Bank)				New	
Cognitive Level	Memory or Fundamental Knowledge			Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describe or recognize relationships				
10CRF55 Content	55.41	10	55.43		
	(SRO Only)				
Time to Complete: 1-2 minutes					

69 **ID: 09-1 NRO69** **Points: 1.00**

The plant was at rated power when a control power fuse blew inside a Control Room Panel. The Shift Manager has declared that the event did **not** create an emergency situation.

The original fuse was labeled as being manufactured by the Littelfuse Corporation. A similar fuse, labeled as being manufactured by the Bussmann Corporation, has been obtained from a controlled fuse location.

PIMS shows that the new fuse is a like-for-like replacement for the blown fuse.

IAW CC-AA-206, Fuse Control, which of the following is correct?

A. **Only** the Craft may install the fuse with **no** further engineering evaluation.

B. The Operator may install the fuse with **no** further engineering evaluation.

C. The Operator may **not** install the fuse since it is **not** an emergency situation.

D. The Operator may install the fuse **only after** the fuse is evaluated by the Fuse Engineer.

Answer: B

Answer Explanation:

QID: 09-1 NRO69		
Question # / Answer	69	Developer/Date: NTP 12/29/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Equipment Control 2.2.14 - Knowledge of the process for controlling equipment configuration or status.					3.9	4.3
Level	RO	Tier	3	Group		
General References	CC-AA-206					

Explanation	The plant was at rated power when a control panel fuse blew. It has been determined that the new fuse is like-for-like with the old fuse. IAW the reference, the Operator may install the fuse and no further engineering evaluation is required. Answer B is correct. Since the Operator is able to install the fuse, answer A is incorrect. IAW the reference, Operators may install fuses in both emergency and non-emergency situations. Answer C is incorrect. Since the fuse has been determined to like-for-like replacement, no further fuse evaluation is required. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps or cautions			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

70 **ID: 09-1 NRO70** **Points: 1.00**

The plant was at rated power when an ATWS occurred.

IAW SP-21, Alternate Insertion of Control Rods, which of the following alternate control rod insertion methods has the potential to **raise** the airborne contamination levels in the Reactor Building?

A. Venting the Scram Air header.

B. Opening the Individual Scram Test Switches.

C. Placing the 100 amp Main RPS Breakers in OFF.

D. Placing the RPS Subchannel Test Keylock switches in TEST.

Answer: B

Answer Explanation:

QID: 09-1 NRO70		
Question # / Answer	**70**	**Developer/Date: NTP 12/30/09**

Knowledge and Ability Reference Information							
K&A						**Importance Rating**	
						RO	**SRO**
Radiation Control **2.3.14 - Knowledge of radiation or containment hazards that may arise during normal, abnormal, or emergency conditions or activities.**						**3.4**	**3.8**
Level	**RO**	**Tier**	**3**	**Group**			
General References	**EMG-SP21**						

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

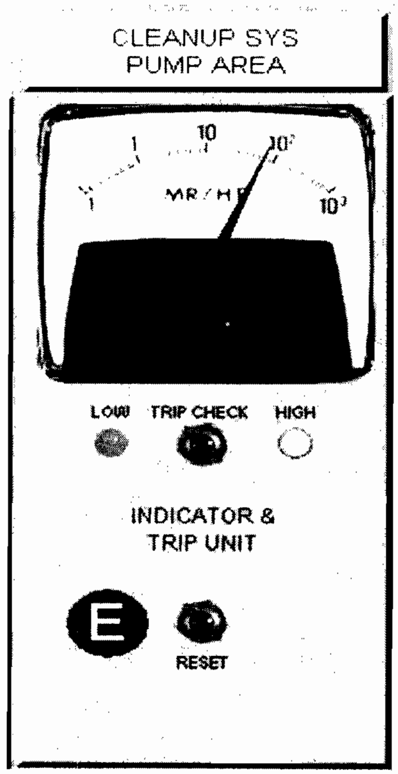
Explanation	The plant was at power when an ATWS occurred. All answers are methods to insert control rods IAW SP-21. When a scram test switch is placed in the scram position, this de-energizes the scram solenoids for the selected control rod. This will allow reactor coolant to travel to the scram discharge volume, which is not isolated, and onto the reactor Building Equipment Drain Tank. On a normal scram, the SDV is isolated from the RBEDT. SP-21 provides a caution while using the scram test panel. Answer B is correct. All answers listed are alternate methods to insert control rods during an ATWS, but none will raise RB contamination levels.		
References to be provided during exam:	**None**		
Learning Objective	2621.845.0.0053 LO 3056A		

Question Source (New, Modified, Bank)			**Modified**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**	**12**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

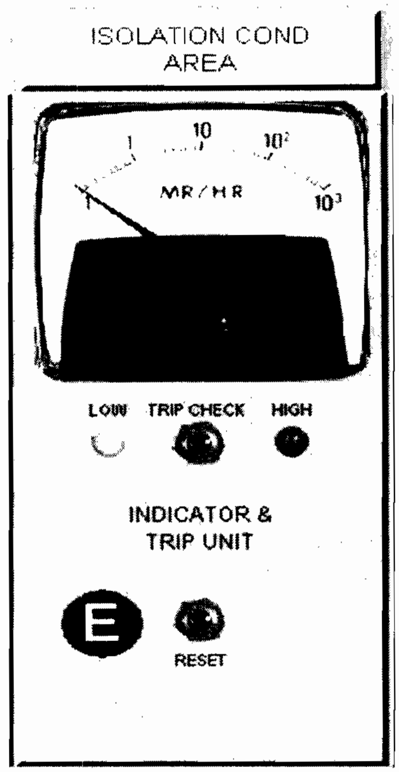
71 **ID: 09-1 NRO71** **Points: 1.00**

Which of the following indications below require entry into an EOP?

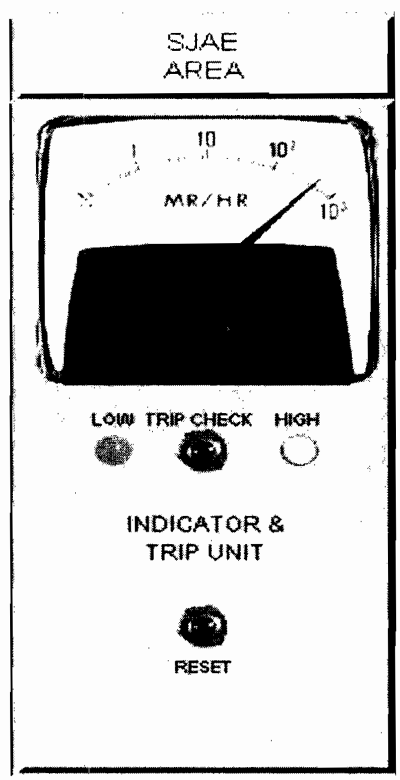
A.



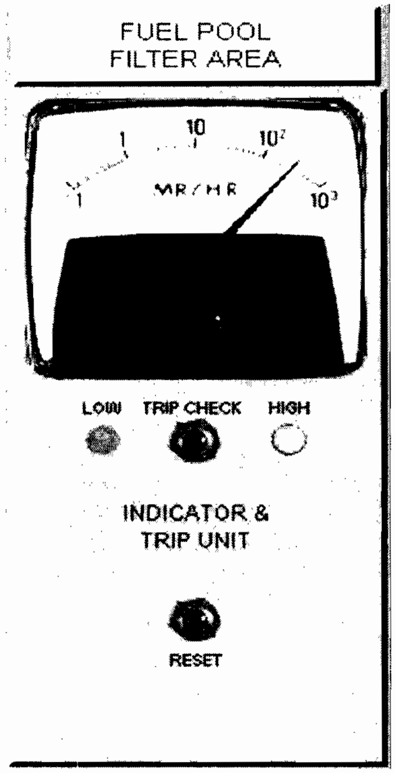
B.


ISOLATION COND
AREA
1
10
10²
MR/HR
10³
LOW
TRIP CHECK
HIGH
INDICATOR &
TRIP UNIT
E
RESET

C.


SJAE
AREA
1
10
10²
MR/HR
10³
LOW
TRIP CHECK
HIGH
INDICATOR &
TRIP UNIT
RESET

D.



Answer: A

Answer Explanation:

QID: 09-1 NRO71		
Question # / Answer	71	Developer/Date: NTP 12/30/09

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
Radiation Control 2.3.5 - Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.				2.9	2.9
Level	RO	Tier	3	Group	
General References	Secondary Containment Control EOP				

Explanation	IW the reference, the Cleanup System Pumps Area ARM (ARM C-1), when above the high alarm setpoint, requires entry into the Secondary Containment Control EOP. Answer A is correct. If the Isolation Condenser area ARM were indicating high, then it would require entry into the Secondary Containment Control EOP. But the ARM is indicating low - not high, then entry into the EOP is not required. Answer B is incorrect. The SJAE ARM in answer C is high, but the ARM is not located in the Secondary Containment and is thus not an EOP entry. Answer C is incorrect. The Fuel Pool Filter Area ARM sounds like it is located inside the Secondary Containment since the Fuel Pool is located there, but the filter is located outside of the Secondary Containment. It also shows that this ARM is high. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.845.0.0057 LO 1667	

Question Source (New, Modified, Bank)				**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**		
	NUREG 1021 Appendix B: Procedure steps and cautions				
10CRF55 Content	**55.41**	**11**	**55.43**		
	(SRO Only)				
Time to Complete: 1-2 minutes					

72 **ID: 09-1 NRO72** **Points: 1.00**

The plant was shutdown for an outage when the fire detection in the EDG 1 enclosure caused a single fire alarm on the Main Fire Alarm Panel A. **No** other annunciators alarmed.

IAW ABN-29, Plant Fires, which of the following is the **first** Control Room response to the event?

A. Sound the Station Fire alarm for 10 seconds.

B. Evacuate personnel from the area of the alarm.

C. Dispatch an Operator to the area of the alarm.

D. Dispatch the Fire Brigade to the area of the alarm.

Answer: C

Answer Explanation:

QID: 09-1 NRO72		
Question # / Answer	**72**	**Developer/Date: NTP 12/60/09**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
Emergency Procedures/Plans G2.4.25 - Knowledge of fire protection procedures.					**3.3**	**3.7**
Level	**RO**	**Tier**	**3**	**Group**		
General References	**ABN-29**					

Explanation	The plant was shutdown when the control room received a fire alarm on the main fire alarm panel. No other alarms came in, which could indicate fire suppression system activation. Because of this, the fire is not confirmed, and the correct first action for a non-confirmed fire is to dispatch an operator to the area. Answer C is correct. The first action for a confirmed fire is to sound the fire alarm for 10 seconds, then announce the location of the fire and request the Fire Brigade to respond over the PA system. All other answers are incorrect but plausible.
References to be provided during exam:	**None**
Learning Objective	2621.828.0.0019 LO 286-10445

Question Source (New, Modified, Bank)		**Bank**		
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**	**10**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

73 **ID: 09-1 NRO73** **Points: 1.00**

The plant was at rated power when a manual scram was inserted IAW ABN-1, Reactor Scram. The Operator makes the following report:

- All RPS GROUP Scram lights are energized
- Reactor power indicates 100%
- Drywell pressure is 4 psig and rising slowly

Over the next three minutes, the Operatots report the following:

- ADS has been bypassed.
- ATWS actions IAW OP-OC-101-111-1001, Stratagies for Successful Transient Mitigation, are complete. These actions have resulted in **all** control rods fully inserting.

Which of the following states the correct plant indications from the events described above?

1. The REACTOR MODE SELECTOR switch is in REFUEL
2. The ROPS BYPASSED annunciator is in alarm
3. The ARI INITIATED annunciator is in alarm
4. The EMRV NORMAL/DISABLE switches are in DISABLE

A. 1 and 2

B. 2 and 3

C. 3 and 4

D. 4 and 1

Answer: B

Answer Explanation:

QID: 09-1 NRO73		
Question # / Answer	73	**Developer/Date: NTP 12/30/09**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

Emergency Procedures/Plan G2.4.46 - Ability to verify that the alarms are consistent with the plant conditions.					**4.2**	**4.2**
Level	**RO**	**Tier**	**3**	**Group**		
General References	**OP-OC-101-111-1001**	**ABN-1**				
Explanation	The plant was at power when an event occurred requiring a manual scram and a full power ATWS was diagnosed. ABN-1 requires that the Reactor Mode Selector switch be placed in Shitdown and both RPS scram button depressed. OP-OC-101-111-1001 directs that for an ATWS condition, the ATC perform the following actions: initiate ARI (alternate rod injection), Bypass ROPs (Reactor Overfill Protection), and reduce recirculation flow to minimum. When ROPs is bypassed, the ROPS BYPASSED annunciator will enwergize. Because the control rods fully inserted due to ARI initiation, the ARI INITIATED annunciator will also be energized. Answer B is correct (selection 2 and 3). The Reactor Mode Selector switch is initially placed in Shutdown IAW ABN-1. But it is not uncommon to place the Mode switch in Refuel during an ATWS if control rod insertion with RMCS is required (manually driving control rods). Since all control rods are inserted, then driving rods is not required and the Mode switch will still be in Shutdown. Selection 1 is incorrect. ADS is bypassed by placing the 2 ADS Timer switches in Bypass. But, the ADS valves can also be bypassed by placing the EMRV Normal/Defeat switches in Defeat. These actions is may be used to close open ADS valves (which will also byp[ass the ADS function when in Defeat). Selection 4 is incorrect. Thus, any answer with a 1 and 4 is incorrect (answers A, C & D).					
References to be provided during exam:	**None**					
Learning Objective	2621.845.0.0012 LO 263-10445					

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**

	NUREG 1021 Appendix B: Predict an event or outcome				
	10CRF55 Content	**55.41**	**10**	**55.43**	
		(SRO Only)			
	Time to Complete: 1-2 minutes				

74 **ID: 09-1 NRO74** **Points: 1.00**

The plant was at rated power when an event occurred. A summary of the event is provided below:

- Total seal leak of Recirculation Pump C
- Recirculation Pump C was shutdown
- Recirculation Pump C PUMP SUCTION Valve V-37-31 switch was placed in CLOSE but **both** the red and green lights remained energized
- The SRO directed venting the Torus through Reactor Building normal ventilation IAW SP-31, Venting the Primary Containment to Maintain Pressure Below 3.0 psig

Five minutes after the venting process began, the following annunciator alarmed:

- STACK EFFLUENT HI

Chemistry has confirmed the elevated radiation readings are from the Primary Containment.

IAW the above RAP, which of the following is required to continue venting?

A. Vent the Primary Containment through the Standby Gas Treatment System.

B. Primary Containment venting may continue in the current lineup since it's directed by the EOPs.

C. Place the keylock DRYWELL VENT-PURGE INTERLOCK BYPASS switch in BYPASS.

D. Place the keylock CNTMT VENT AND PURGE ISOLATION BYPASS switch in BYPASS.

Answer: A

Answer Explanation:

QID: 09-1 NRO74		
Question # / Answer	74	**Developer/Date: NTP 12/30/09**

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

Knowledge and Ability Reference Information							
K&A						Importance Rating	
						RO	SRO
Radiation Control 2.3.11 - Ability to control radiation releases						3.8	4.3
Level	RO	Tier	3	Group			
General References	EMP-SP31		RAP-10F2d				
Explanation	The plant was at rated power when a recirculation pump seal failed (resulting in elevated Drywell pressures and temperatures) and the pump isolation was not complete. The primary Containment Control EOP was entered and Primary Containment venting through the RB normal HVAC system was initiated. The Stack Hi alarm came in and was confirmed to be from the Primary Containment. IAW RAP-19F2d, vent the primary Containment through the Standby Gas treatment System, which is performed in SP-31. Venting through STGS will filter the effluent prior to discharge to the stack. Answer A is correct. Even though the venting is directed IAW the EOPs, SP-31 does also provide venting through the SGTS, which would not be in conflict with the RAP. Answer B is incorrect. The switches in answer C & D sound very close. If the switch in answer D is placed in bypass, this would override the Drywell high pressure isolation of Drywell valves. This would only be done if venting would prevent Primary Containment failure. The switch in answer C is used when inerting and the reactor mode switch is to be placed in RUN. Answers C & D are plausible but incorrect.						
References to be provided during exam:	EMG-SP31						
Learning Objective	2621.845.0.0056 LO 200-10450						

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge	X 1:P	Comprehension or Analysis	
	NUREG 1021 Appendix B: Procedure steps and cautions			

EXAMINATION ANSWER KEY

ILT 09-1 NRC RO Exam

10CRF55 Content	55.41	12	55.43	
	(SRO Only)			
Time to Complete: 1-2 minutes				

75 **ID: 09-1 NRO75** **Points: 1.00**

The plant was at 900 psig during a startup when an event occurred. The SRO directed entry into 2 EOPs from a single common parameter.

Which of the following states the plant response from the event?

1. The Core Spray System automatically initiated
2. The Startup Breakers automatically closed
3. The Emergency Diesel Generators automatically fast started
4. The Standby Gas Treatment System automatically initiated

A. 1 and 2.

B. 2 and 3.

C. 1 and 4.

D. 3 and 4.

Answer: C

Answer Explanation:

<table>
<tr><td colspan="3">QID: 09-1 NRO75</td></tr>
<tr><td>Question # / Answer</td><td>75</td><td>Developer/Date: NTP 12/31/09</td></tr>
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<tr><th colspan="8">Knowledge and Ability Reference Information</th></tr>
<tr><th colspan="6" rowspan="2">K&A</th><th colspan="2">Importance Rating</th></tr>
<tr><th>RO</th><th>SRO</th></tr>
<tr><td colspan="6">Emergency Procedures/Plans
2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.</td><td>4.5</td><td>4.6</td></tr>
<tr><td>Level</td><td>RO</td><td>Tier</td><td>3</td><td>Group</td><td></td><td colspan="2"></td></tr>
<tr><td>General References</td><td colspan="2">EMG-SP1
2621.828.0.0042</td><td colspan="3">RPV Control - No ATWS EOP
2621.828.0.0013</td><td colspan="2">Primary Containment Controp EOP</td></tr>
</table>

Explanation	The plant was at 900 psig during a startup when an event occurred and 2 EOPs were entered from a single common parameter. The only single common parameter which would require entry into 2 EOPs is Drywell high pressure (RPV Control - No ATWS and primary Containment Control EOPs). The following occur on a Drywell high pressure condition: reactor scram, Primary Containment isolation, auto start of Standby Gas, auto idle start of the emergency diesel generators, and core spray auto start. Of these, only selections 1 and 4 are correct. Answer C is correct. In the conditions provided, the Startup breakers are already closed and will not auto close (selection 2). Had the event occurred from rated power, then the startup breakers would auto close following the scram and turbine trip. Answer A is incorrect. The diesel generators will fast start from a loss of voltage condition, but only idle start on a LOCA signal (Drywell high pressure or RPV water level lo-lo). Answer B is incorrect for this reason and from what has already been stated about the Startup breakers.. Answer D is incorrect because of what has been stated about the diesel generators. 4/10/10: This question was modified following NRC feedback.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0010 LO 209-10445	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**	**7**	**55.43**	
	(SRO Only)			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

1	ID: 09-1 NSRO1	Points: 1.00

The plant was at rated power preparing for a shutdown due to an overload trip of FEEDER BREAKER 1A2P.

An event then occurred. Present plant conditions include the following:

- Most control rods indicate full-in with several control rods at various positions between 24 and 44
- Reactor power is slowly oscillating between 3% and 5% on APRMs, with LPRMs oscillating 5 watts/cm^2 peak-to-peak
- RPV water level indicates 80" and lowering slowly
- Drywell pressure indicates 19 psig and rising slowly
- Drywell temperature indicates 279 ^{o}F and rising slowly
- Containment Spray System 2 DRYWELL SPRAY DISCHARGE V-21-5 is stuck closed
- Drywell OXYGEN CONCENTRATION indicates 3.4%
- Torus water temperature indicates 92 ^{o}F and rising very slowly

Which of the following states the **next** SRO direction?

A. Initiate the Standby Liquid Control System IAW SP-22, Initiating the Liquid Poison System.

B. Emergency Depressurize the RPV IAW the Emergency Depressurization - With ATWS EOP.

C. Initiate Torus Cooling IAW SP-25, Initiation of the Containment Spray System in the Torus Cooling Mode.

D. Lineup and initiate Drywell Sprays IAW SP-29, Initiation of the Containment Spray System for Drywell Sprays.

Answer: B

Answer Explanation:

QID: 09-1 NSRO1		
Question # / Answer	1	**Developer/Date: NTP 12/31/09**

Knowledge and Ability Reference Information	
K&A	**Importance Rating**

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

	RO	SRO
295028 High Drywell Temperature EA2.02 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Reactor pressure		3.9

Level	SRO	Tier	1	Group	1

General References	Primary Containment Control EOP	RPV Control - With ATWS EOP	EOP Users Guide

Explanation	The plant was at rated power with USS Bus 1A2 lost. Drywell temperature is 279 F and rising. The Primary Containment Control EOP states that when it has been determined that bulk Drywell temperature cannot be maintained below 281 F, then ED is required. The question stem states that power is 3-5% on APRMs and several rods are not fully inserted. Drywell sprays are designed to reduce Drywell temperature and pressure in cases of elevated Drywell temperatures/pressures such as this. But, with USS 1A2 de-energized (which normally powers Containment Spray Pumps A & B), and the stuck valve V-21-5 (which prevents Drywell Sprays in Containment Spray System 2), there are no containment sprays available while Drywell temperature is rising. Thus, ED - With ATWS is required. Answer B is correct. Starting Standby Liquid Control is not a correct action in this ATWS condition. The RPV Control - With ATWS directs starting SLC in 2 cases: 1)periodic oscillations on LPRMs exceed 30 watts/cm^2 peak-to-peak; 2) When Torus water temperature cannot be maintained less that the BIIT Curve. The most restrictive Torus water temperature is 110 F (at $\geq$ 10% power). At the current reactor power level, the BIIT limit is > 130 F. Thus, under the given conditions, initiating SLC is not appropriate. Answer A is incorrect. The Primary Containment Control EOP says to maintain Torus water temperature below 95 F, and to place a loop of Torus cooling in service. Under the given conditions, Torus water temperature is being maintained with no further actions. And, comparing the actions to ED or Torus cooling, then Torus cooling is of lower importance and thus not the first action. Answer C is incorrect. As stated previously, it would be beneficial to be spraying the Drywell to lower temperatures and Pressures. If a Drywell Spray system were available, then initiating it to lower Drywell parameters to prevent an ED would be a viable option. But since no Drywell Sprays are available, then Answer D is incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.845.0.0053 LO 3055A	

<table>
<tr><td colspan="3">Question Source (New, Modified, Bank)</td><td colspan="2">New</td></tr>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X
3:SPK</td></tr>
<tr><td colspan="4">NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td></td><td>55.43</td><td>5</td></tr>
<tr><td colspan="4">(SRO Only) Assessment of conditions and selection of appropriate procedure</td></tr>
<tr><td colspan="5">Time to Complete: 1-2 minutes</td></tr>
</table>

2 **ID: 09-1 NSRO2** **Points: 1.00**

The plant was preparing to initiate a shutdown from 83% power after being on-line for 297 days, when a catastrophic loss of instrument air occurred. Present plant conditions include the following:

- INST AIR SUPPLY PRESS indicates 0 psig
- **All** control rods indicate fully inserted **except** 18-15 and 18-19 which indicate position 48, with their individual red scram lights de-energized
- **All** LPRM amber lights on Panel 4F are energized
- RPV water level is 128" and rising
- RPV pressure is 1018 psig and rising

Which of the following shall the SRO direct?

A. Using RMCS, manually insert control rods 18-15 and 18-19 IAW SP-21, Alternate insertion of Control Rods.

B. Control RPV pressure using the Isolation Condensers IAW SP-11, Alternate Pressure Control Systems - Isolation Condensers.

C. Maintain RPV water level 138" - 175" controlling with the Feedwater MFRVs IAW SP-19, Feedwater/Condensate and CRD System Operation.

D. Augment RPV pressure control by operating RWCU in the Letdown Mode IAW SP-14, Alternate Pressure Control Systems, Clean-up in Letdown Mode.

Answer: B

Answer Explanation:

QID: 09-1 NSRO2		
Question # / Answer	2	**Developer/Date: NTP 1/2/10**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

295019 Partial or Total Loss of Inst. Air AA2.01 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR: Instrument air system pressure		**3.6**

Level	**SRO**	**Tier**	**1**	**Group**	**1**

General References	**RPV Control - w/ATWS EOP**	**EMG-SP21 EMG-SP11**	**EMG-SP19 EMG-SP14 ABN-35**

Explanation	The plant was at power when catastrophic loss of instrument air occurred. From this, outboard MSIVs will close and control rods will scram from MSIV position. 2 control rods failed to scram and are at position 48 and the ATWS EOP is entered. Since the MSIVs are closed, the Isolation Condensers are available for use to control RPV pressure IAW SP-11. Answer B is correct. SP-21 does allow manual control rod insertion during an electric ATWS, but with instrument air gone, both CRD FCVs auto close. With these closed, there is no driving water flow to manually insert the control rods. Answer A is incorrect. The ATWS EOP does direct an RPV water level of 138"-175" using SP-19, but with a loss of instrument air, the MFRVs lockup (but may drift open/closed) and thus the valves cannot be controlled from the control room. Answer C is incorrect. The RPV Control - w/ATWS EOP allows use of the RWCU in the letdown mode, but with a loss of instrument air, RWCU has isolated and the letdown FCV ND22 has failed closed. Answer D is incorrect.

References to be provided during exam:	**None**	

Learning Objective	2624.845.0.0053LO 3055A

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			

10CRF55 Content	55.41		55.43	5
	(SRO Only) Assessment of facility conditions and selection of appropriate procedures			
Time to Complete: 1-2 minutes				

3 **ID: 09-1 NSRO3** **Points: 1.00**

The plant was at rated power when the following annunciator alarmed:

- RECIRC PUMP A – CCW FLOW LO A (E7d)

The Operator reports the following indications:

- Recirculation Pump A SEAL CAVITIES PRESSURE NO. 1 indicates 1020 psig and steady
- Recirculation Pump A SEAL CAVITIES PRESSURE NO. 2 indicates 1020 psig and steady
- UNIDENTIFIED DRYWELL LEAKAGE has risen by 1.9 GPM
- Drywell temperature and pressure remain **unchanged**

Two minutes later, the Operator reports the following:

- Recirculation Pump A Cavity Temperatures indicate a rising trend

Which of the following actions shall the SRO direct?

A. Trip and isolate Recirculation Pump A due to the failure of the NO. 2 seal IAW ABN-2, Recirculation System Failures.

B. Reduce unidentified leakage to within allowable limits within 8 hours or place the reactor in the shutdown condition IAW TS 3.3, Reactor Coolant.

C. Due to an RBCCW leak, trip Recirculation Pump A when cavity temperatures rise to the values specified IAW ABN-19, RBCCW Failure Response.

D. In order to minimize the seal cavity heatup, place Recirculation Pump A in MAN and reduce Recirculation Pump A speed IAW 302.1, Reactor Recirculation System.

Answer: C

Answer Explanation:

QID: 09-1 NSRO3		
Question # / Answer	**3**	**Developer/Date: NTP 1/4/10**

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

<table>
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<tr><td colspan="4" rowspan="2">K&A</td><td colspan="2">Importance Rating</td></tr>
<tr><td>RO</td><td>SRO</td></tr>
<tr><td colspan="4">295018 Partial or Total Loss of CCW
AA2.03 - Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Cause for partial or complete loss</td><td></td><td>3.5</td></tr>
<tr><td>Level</td><td>SRO</td><td>Tier</td><td>1</td><td>Group</td><td>1</td></tr>
<tr><td>General References</td><td>ABN-19</td><td colspan="2">RAP-E7d</td><td colspan="2">ABN-2</td></tr>
</table>

Explanation	The plant was at rated power when a low cooling water flow/loss of cooling water flow event to Recirculation Pump A is annunciated. When this applies a single recirculation pump, the associated RAP directs that the actions in ABN-19, RBCCW Failure Response, apply. When either seal cavity temperatures, motor bearing temperatures or motor winding temperature limits are reached, then ABN-19 requires tripping the pump and referring to ABN-2, Recirculation System Failures. Conditions also show that Drywell unidentified leakage has risen by 1.9 GPM, but Drywell temperature, pressure and humidity show no change. Thus, the cause of the increased leak rate cannot be attributed to reactor coolant, but can be from RBCCW. Thus, a leak from RBCCW into the Drywell is the cause for both the low flow alarm and the rise in unidentified leakage. Answer C is correct. The question stem also shows a problem with the recirculation pump seals. The indications show a failure of the No. 1 seal, and actions IAW ABN-2 apply. These actions include removing the pump from service and evaluating isolating the recirculation loop. Answer A provides these actions, but due to a failure of the No. 2 seal - not the No. 1 seal. Thus, answer A is incorrect. TS 3.3, Reactor Coolant, states a limit of 2 gpm increase in unidentified leakage within a 24 hour period. If exceeded, it shall be reduced to within limits in 8 hours, or place the reactor in a shutdown condition in the next 12 hours. Since the increase in unidentified leakage is only 1.9 gpm, the specification does not apply and answer B is incorrect. Generally, when cooling to a component is reduced, good engineering practice is to reduce load on the component to reduce the heatup of the component. But, ABN-2, says in a note that recirculation pump speed changes should be minimized as much as possible, and does not provide any direction to reduce pump speed. Answer D is incorrect. 4/9/10: This question was modified following NRC feedback.

References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0038 LO 202-10445	

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<tr><td colspan="3">Question Source (New, Modified, Bank)</td><td colspan="2">New</td></tr>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X
2:DR</td></tr>
<tr><td colspan="4">NUREG 1021 Appendix B: Describe or recognize relationships</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td></td><td>55.43</td><td>5</td></tr>
<tr><td colspan="4">(SRO Only) Assessment of facility conditions and selection of appropriate procedures</td></tr>
<tr><td colspan="5">Time to Complete: 1-2 minutes</td></tr>
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4 **ID: 09-1 NSRO4** **Points: 1.00**

The plant was at 425 psig during a startup when the following annunciator alarmed:

- TORUS LEVEL HI/LO

5 minutes later, the following reports were made:

- A large Torus leak has developed inside the Torus Room
- Both TORUS LEVEL WIDE RANGE indicators show 120" and lowering
- The STA reports that Torus water level will be < 90"" by the time that Torus makeup is injecting IAW SP-37, Makeup to the Torus via Core Spray System

IAW the EOP Users Guide, which of the following states the **next** required action and the bases for the action?

	Required Action	Basis
A.	Initiate a reactor shutdown as directed in the TORUS LEVEL HI/LO alarm response	This allows a normal RPV depressurization before the pressure suppression function of the Primary Containment is lost
B.	Emergency Depressurize the RPV as directed in the Primary Containment Control EOP	This allows a rapid RPV depressurization before the pressure relief function of the EMRVs is lost
C.	Scram as directed in the Primary Containment Control EOP	This allows a rapid RPV depressurization with Turbine Bypass Valves which reduces the burdon on the Torus suppression capability
D.	Scram as directed in the Primary Containment Control EOP	This allows a rapid RPV depressurization with Turbine Bypass Valves before losing NPSH to the Core Spray Pumps

Answer: C

Answer Explanation:

QID: 09-1 NSRO4		
Question # / Answer	**4**	**Developer/Date: NTP 1/4/10**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
295006 SCRAM 2.4.18 - Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.						**4.0**
Level	**SRO**	**Tier**	**1**	**Group**	**1**	
General References	**EOP Users Guide**					
Explanation	The plant was at 425 psig when a Torus water level hi/lo alarm came in. 5 minutes later, Torus water level is 120" and lowering, and the STA has determined that Torus water level will be below 90" before Torus makeup injection will start. Therefore, it can be determined that Torus water level will go below 110", and the Primary Containment Control EOP requires a manual scram. At <110', the downcomers become uncovered and steam from any LOCA will directly pressurize the Torus air space instead of being forced and quenched in the Torus water. IAW the reference, reducing RPV pressure through the Turbine Bypass valves reduces the burden on the Torus syppression capability, which is already diminished due to the low water level. Answer C is correct. The basis for answers A & B are logical, as both of these can occur if Torus water level continues to lower. The very next step in the Primary Containment Control EOP, after scram of the reactor, is to Emergency Depressurize. But, a manual scram is to take place first. Answers A & B are incorrect. Answer D also has a logical basis, but it is not the basis for the step to scram. Answer D is incorrect.					
References to be provided during exam:	**None**					

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

Learning Objective	2621.845.0.0056 LO 200-10445

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:B**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Basis or purpose			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility conditions and selection of appropriate procedure			
Time to Complete: 1-2 minutes				

5 **ID: 09-1 NSRO5** **Points: 1.00**

The plant was at rated power when a series of events occurred. The time line of the events is provided below:

0800 The Control Room receives notification of a fire in the Reactor Building
0801 ABN-29, Plant Fires, is entered
0805 Annunciator ROD DRIFT alarms
0805 The Operator reports control rod 26-27 is drifting outward
0806 ABN-6, Control Rod Malfunctions, is entered
0814 The Fire Brigade Leader reports the fire has been extinguished
0815 Annunciator OFFGAS HI alarms
0816 ABN-26, High Main Steam/Off-Gas/Stack Effluent Activity, is entered
0830 Annunciator OFFGAS HI-HI alarms
0831 The Operator reports that STACK EFFLUENT High Range Monitor indicates 2 μCi/cc and rising slowly
0845 The Operator reports that Offgas has successfully isolated
0846 The Operator reports that **no** ARMs in the Turbine Building or Reactor Building have reached the MAX SAFE value
0847 The Operator reports that STACK EFFLUENT High Range Monitor indicates 3 μCi/cc and lowering slowly

Which of the following states the correct emergency plan declaration?

A. Alert (Radiological Effluent)

B. Alert (Abnormal Rad Levels)

C. Unusual Event (Fire/Explosion)

D. Unusual Event (Abnormal Rad Levels)

Answer: A

Answer Explanation:

QID: 09-1 NSRO5		
Question # / Answer	5	**Developer/Date: NTP 1/4/10**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

295038 High Offsite Release Rate **2.4.11 - Emergency Procedures / Plan: Knowledge of abnormal condition procedures.**							**4.2**
Level	**SRO**	**Tier**	**1**	**Group**	**1**		
General References	**ABN-29** **EP-AA-1010**		**ABN-2**		**ABN-26**		
Explanation	The plant is at rated power when several events occur: a fire, and a control rod drift which results in fuel failures, and a rise on offgas activity and stack activity. It is shown that the Stack activity exceeds the Alert level at time 0831. At 16 minutes later, it shows that the Stack activity, although lowering, is still above the alert level. The radiation level must be above the Alert Table for ≥ 15 minutes. Emergency classification for the alert (RA1, Radiological Effluent) is correct. Answer A is correct. Radiation readings > 2000 mr/hr in areas of the Turbine or Reactor Building would require an Alert emergency classification (RA3). The question states that no RB or TB ARM has reached the Max Safe value, which is 1000 mr/hr. Therefore, answer B is incorrect. A UE for a fire requires that the fire in the reactor Building not extinguished within 15 minutes. The question shows that the reactor Building fire is extinguished in 14 minutes. Answer C is incorrect. Unusual Event RU3 classification is required if offgas isolation occurs or should occur from a valid offgas radiation monitor signal, which has occurred. As a minimum, this UE would be declared if it were the only classification. Answer D is incorrect.						
References to be provided during exam:	**Hot Matrix from EP-AA-1010**						
Learning Objective	G-101 DBIG LO G-101 DBIG-01						

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X** **3:SPR**
	NUREG 1021 Appendix B: Solve a problem with a reference			

10CRF55 Content	**55.41**		**55.43**	**1**
	(SRO Only) Assessment of facility conditions and selection of appropriate procedures			
Time to Complete: 1-2 minutes				

6 | **ID: 09-1 NSRO6** | **Points: 1.00**

The plant was at rated power when an automatic scram setpoint was exceeded. Present plant conditions include the following:

- Generator load indicates 135 MWe
- The REACTOR MODE SELECTOR switch is in SHUTDOWN
- The MASTER RECIRCULATION SPEED CONTROLLER indicates 35 hertz
- ROPS is in BYPASS
- ALT ROD INJECTION SYS has been initiated
- RPV water level indicates 127" and rising
- SP-1, Confirmation of Automatic Initiation and Isolations, is being performed

Which of the following states the **next** SRO EOP direction and basis for this direction IAW the EOP Users Guide?

	EOP Direction	EOP Basis
A.	Bypass the RPV Lo-Lo water level MSIV closure	Prevents loss of the primary heat sink and the potential impact on the Primary Containment
B.	Bypass the RPV low pressure MSIV closure	Prevents loss of the primary heat sink and the potential impact on the Primary Containment
C.	Trip all Recirculation Pumps	Reduces reactor power and minimizes the potential for power oscillations
D.	Inhibit ADS by placing all EMRV keylock switches to DISABLE	Prevents an unnecessary RPV depressurization and injection of cold unborated water into the RPV

Answer: A

Answer Explanation:

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

QID: 09-1 NSRO6		
Question # / Answer	6	Developer/Date: NTP 1/5/10

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
295037 SCRAM Conditions Present and Reactor Power Above APRM Downscale or Unknown 2.4.20 - Emergency Procedures / Plan: Knowledge of operational implications of EOP warnings, cautions, and notes.						4.3
Level	SRO	Tier	1	Group	1	
General References	EIOP Users Guide					

Explanation	The plant was at power when an auto scram setpoint occurred and a high power ATWS (>2% power) occurred. The next action in the ATWS EOP level/power leg is to bypass the RPV lo-lo water level MSIV closure. The caution associated with this step says that isolation of the MSIVs under ATWS conditions may result in Primary Containment failure. In the following step, because power is >2%, is to lower RPV water level which could result in closure of the MSIVs and will require the use of the EMRVs which discharges into the Primary Containment. Answer A is correct. Answer B is incorrect since the RPV low pressure MSIV closure is already bypassed when the Reactor Mode Switch is taken to shutdown. Keeping the MSIVs open does ensure the primary heat sink (condenser) stays available. Answer B is incorrect. In the power leg, it asks if the generator is still on-line. If yes, the action is to runback flow to minimum, and then trip the recirculation pumps if power is > 2%. This will reduce power and prevent a turbine trip during the RPV water level excursion. It can be seen that at 35 Hz, recirculation flow is not yet at minimum (about 11 Hz). Thus, running recirculation flow to minimum is performed prior to tripping the recirculation pumps. Answer C is incorrect. Defeating ADS is also a correct next action, but the method of performing this is incorrect in answer D. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.845.0.0053 LO 200-10445A	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only)			
Time to Complete: 1-2 minutes				

7 **ID: 09-1 NSRO7** **Points: 1.00**

The plant was at rated power with the following abnormal lineup:

- Core Spray Main Pump NZ01D was tagged out of service as of 0800 this morning

A grid disturbance occurred which resulted in the following annunciator at 1100:

- LKOUT RELAY 86/S1B TRIP

IAW Tech Specs and procedure OP-OC-108-104-1001, Guidance for Limiting and Administrative Conditions for Operations, which of the following states the required action?

A. The reactor shall be placed in cold shutdown within the next 30 hours.

B. The reactor shall be placed in cold shutdown within 7-days from the pump inoperability.

C. The reactor shall be placed in cold shutdown within 15-days from the pump inoperability.

D. The reactor shall be placed in cold shutdown within 7 days from receipt of the above annunciator.

Answer: A

Answer Explanation:

QID: 09-1 NSRO7		
Question # / Answer	**7**	**Developer/Date: NTP 1/5/10**

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
700000 Generator Voltage and Electric Grid Disturbances 2.2.40 - Equipment Control: Ability to apply technical specifications for a system.					4.7
Level	SRO	Tier	1	Group	1

General References	TS 3.7	OP-OC-108-104-1001	341
Explanation	The plant was at rated power with Core Spray Main Pump NZ01D out of service (powered from 4160 Bus 1C, which can be supplied from Startup transformer S1A). A grid disturbance then results in the loss of Startup Transformer S1B. Procedure 341 states that Core Spray Pumps are engineered safeguard loads. TS 3.7.B provides the following: The reactor shall be PLACED IN the COLD SHUTDOWN CONDITION if the availability of power falls below that required by Specification A above, except that 1. The reactor may remain in operation for a period not to exceed 7 days if a startup transformer is out of service. None of the engineered safety feature equipment fed by the remaining transformer may be out of service. If just the Startup Transformer were lost, a 7-day LCO would be correct. But since the Core Spray Pump NZ01D is powered from the remaining transformer, the plant shall be placed in cold shutdown. IAW OP-OC-108-104-1001, it states that the requirement to be in cold shutdown (with no time provided) shall be within 30 hours. Answer A is correct. The TS for Core Spray does allow a 7-day and a 15-day LCO. But since the times listed are incorrect for the given conditions, then answers B & C are incorrect. Answer D is incorrect since the time is incorrect.		

References to be provided during exam:	TS 3.7 (No basis)	
Learning Objective	2621.828.0.0010 LO 209-10451	

Question Source (New, Modified, Bank)				New	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPK	
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning				
10CRF55 Content	55.41		55.43	2	
	(SRO Only) Facility operating limitations in the technical specifications and their bases				
Time to Complete: 1-2 minutes					

8 **ID: 09-1 NSRO8** **Points: 1.00**

The plant was at rated power when an automatic scram setpoint was exceeded. Current plant conditions include the following:

- All control rod position indication is lost
- All individual scram lights (Panel 4F) are energized
- RPV water level is 141" and lowering
- RPV pressure is 990 psig and steady
- Torus water temperature is 94 °F and steady
- Drywell pressure is 9 psig and rising slowly
- Drywell temperature is 201 °F and rising slowly
- TOTAL STEAM FLOW is 2.01 MLB/HR
- All MFRV FLOW CONTROLLERs are in MAN and TOTAL FEEDWATER FLOW indicates 0 MLB/HR
- ISOL SIGNAL BYPASS V-6-395 is in BYPASS

IAW the EOP Users Guide, which of the following states the strategy of **highest** priority?

A. In the Pressure leg of the Primary Containment Control EOP, direct initiating Drywell Sprays.

B. In the Level/Power leg of the RPV Control - With ATWS EOP, direct an RPV water level band to below 30".

C. In the Power leg of the RPV Control - With ATWS EOP, direct inserting control rods by venting the scram air header.

D. In the Torus Water Temperature leg of the Primary Containment Control EOP, initiating Containment Spray in the Torus Cooling Mode.

Answer: B

Answer Explanation:

QID: 09-1 NSRO8		
Question # / Answer	**8**	**Developer/Date: NTP 1/5/10**

Knowledge and Ability Reference Information	
K&A	**Importance Rating**

	RO	SRO
295009 Low Reactor Water Level **AA2.02 - Ability to determine and/or interpret the following as they apply to LOW REACTOR WATER LEVEL : Steam flow/feed flow mismatch**		**3.7**

Level	**SRO**	**Tier**	**1**	**Group**	**2**
General References	**EOP Users Guide**				
Explanation	The plant was at power when an ATWS occurred. Control rod position has been lost and no direct power indication is provided. But with steam flow at 2 Mlb/hr, and steady RPV pressure, power is about 28% (rated steam flow is about 7.1 Mlb/hr). The question stem also shows that the MSIV lo-lo isolation bypass (SP-16) has been performed (V-6-395 in bypass). Because reactor power is so high, the next task is to terminate/prevent injection IAW SP-17 by stopping all Feedwater Pumps. This also has been performed and accounts for the large steam/feedwater flow mismatch. The next direction is to direct an RPV water level to below 30" IAW the ATWS EOP. Answer B is correct. Maintaining Primary Containment intact is certainly important, but Drywell Sprays cannot be initiated until Drywell/Torus pressure exceeds 12 psig (and its only 9 psig currently). Therefore, Drywell Sprays cannot be initiated from the pressure leg of the Primary Containment Control EOP. Answer A is incorrect. Inserting control rods is also of high priority, but the method to insert control rods is for an electric ATWS and indications given suggest a hydraulic ATWS. Answer C is incorrect. With Torus water temperature at 93 ^{o}F and steady, the entry condition from this parameter into Primary Containment Control EOP has not been reached. Since temperature is < 95 ^{o}F and is steady, there is no hurry in initiating Torus Cooling and is of lower priority than lowering RPV water level through termination/prevention of RPV injection. Answer D is incorrect. 4/9/10: This question was modified following NRC feedback.				
References to be provided during exam:	**None**				

Learning Objective	2621.845.0.0053 LO 200-10445A

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility conditions and selection of procedure			
Time to Complete: 1-2 minutes				

9 **ID: 09-1 NSRO9** **Points: 1.00**

The plant was at rated power when the following annunciators alarmed:

- ROPS ACTUATE A
- RX LVL HI II

The Operator reports that indication for RE05A in Panel 19R indicates upscale at 185".

1) IAW ABN-59, RPV Level Instrument Failures, which of the following actions is required; and,
2) What action is required by Tech Specs 3.1.1, Protective Instrumentation?

	1) ABN-59 Action	2) TS Action
A.	Place the redundant GEMAC level instrument in control	Within 12 hours, restore the instrument or place the instrument in the Trip Condition
B.	Place RPS 1 Subchannel test Switches to TEST	Within 6 hours, restore the instrument or place the instrument in the Trip Condition
C.	Insert a manual 1/2 scram on RPS 1	Within 24 hours, restore the instrument or place the instrument in the Trip Condition
D.	Confirm all automatic actions have occurred from the failed instrument	Within 12 hours, restore the instrument or place the instrument in the Trip Condition

Answer: D

Answer Explanation:

QID: 09-1 NSRO9		
Question # / Answer	9	**Developer/Date: NTP 1/5/10**

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
295008 High Reactor Water Level 2.1.20 - Conduct of Operations: Ability to interpret and execute procedure steps.						4.6
Level	SRO	Tier	1	Group	2	
General References	ABN-59		TS 3.1.1			

Explanation	The plant was at rated power when RPS RPV water level instrument RE05A failed upscale. This instrument inputs into the turbine trip circuit and Feedwater Pump trip on RPV high water level, and inputs into the reactor scram on RPV low water level. When just one level instrument fails upscale, there are no automatic actions which occur. ABN-59 says that if an instrument malfunction has occurred or is suspected (as compared to a failed indicator), then confirm all automatic actions have occurred due to the failed instrument. TS 3.3.1, note nn provides the following (from the scram on low water level portion of the Table): With one required channel inoperable in one Trip System, within 12 hours, restore the inoperable channel or place the inoperable channel and/or that Trip System in the tripped▲ condition. With two or more required channels inoperable: 1. Within one hour, verify sufficient channels remain OPERABLE or tripped▲ to maintain trip capability, and 2. Within 6 hours, place the inoperable channel(s) in one Trip System and/or that Trip System▲▲ in the tripped condition▲ , and 3. Within 12 hours, restore the inoperable channels in the other Trip System to an OPERABLE status or tripped. Otherwise, take the Action Required. Therefore, the instrument must be placed in the trip condition within 12 hours. Answer D is correct. The ABN action in answer A is correct if the failed instrument input into Feedwater Level Control, but the RE05A does not. The TS action time is correct. Answer A is incorrect. If the candidate confuses a scram setpoint with RPV high water level, then the ABN actions in answers B & C might be correct. But even though it is a scram water level instrument, there is no scram setpoint on RPV water level high. Answers B & C are incorrect. 4/9/10: This question was modified following NRC feedback. Also, added TS title in the stem.

References to be provided during exam:	**Reference Deleted**	
Learning Objective	2621.828.0.0030 LO 1032	

Question Source (New, Modified, Bank)	**New**

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

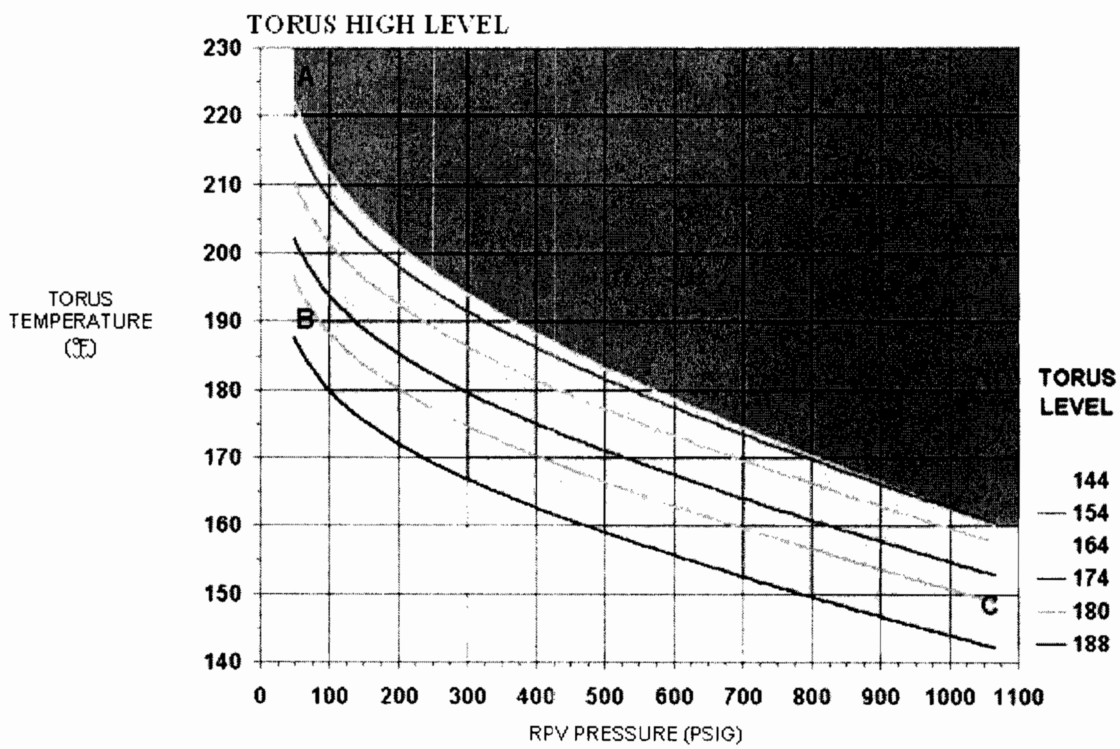
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPR
	NUREG 1021 Appendix B: Solve a problem using a reference			
10CRF55 Content	55.41		55.43	2
	(SRO Only) Facility operating limitations in the technical specifications and their bases			
Time to Complete: 1-2 minutes				

10 **ID: 09-1 NSRO10** **Points: 1.00**

The plant was at rated power when an event occurred. Present plant conditions include the following:

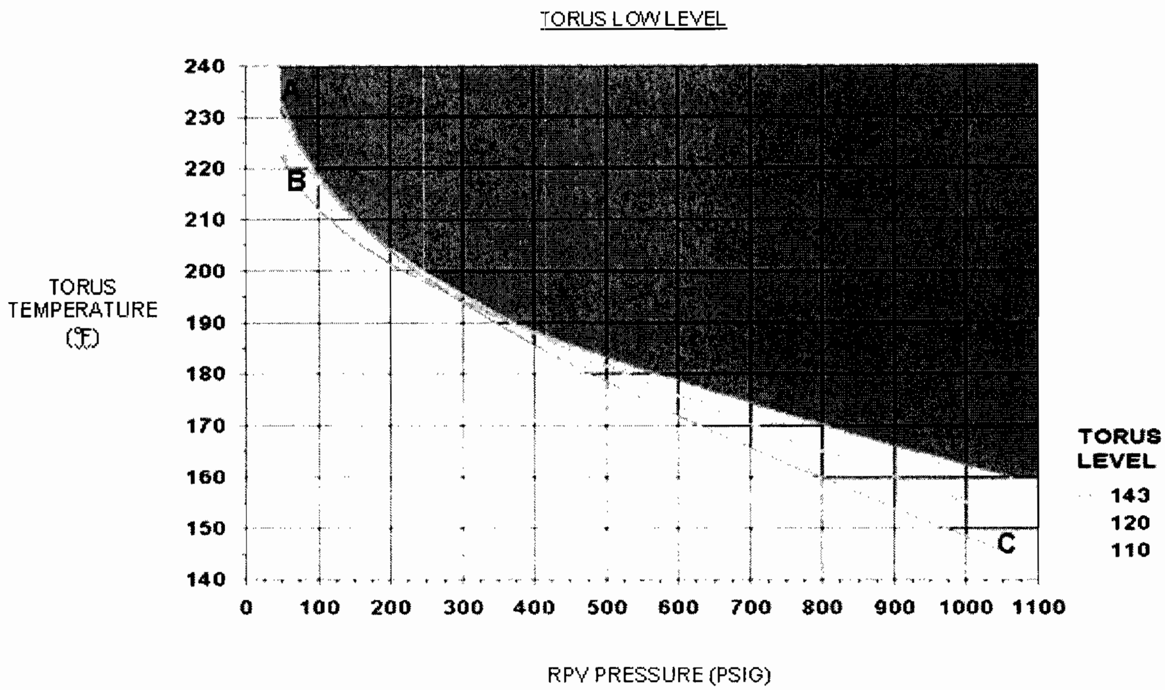
- RPV water level is 80"
- RPV pressure is 700 psig
- Drywell pressure is 12 psig
- Torus pressure is 11 psig
- Drywell temperature is 239 °F
- Torus water level is 164 "
- Torus water temperature is 155 °F

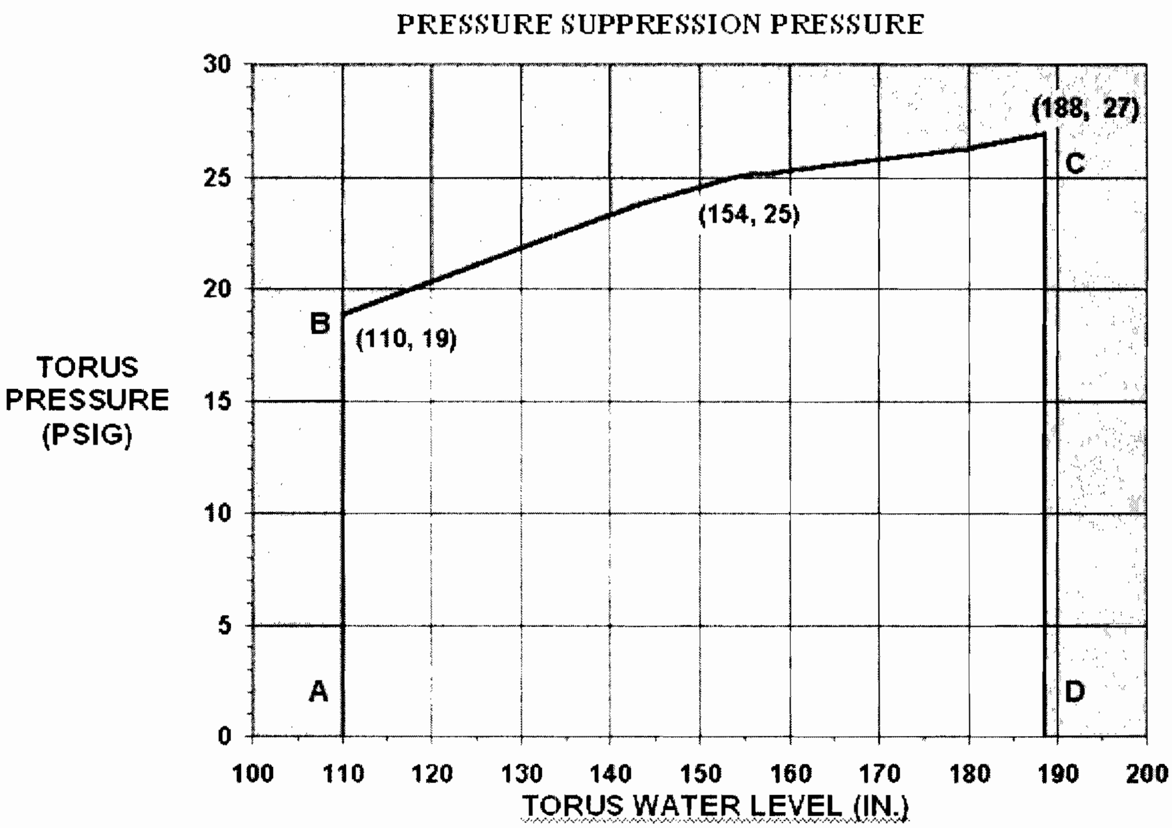
Which of the following is required given the associated parameter change? (Assume all other parameters **remain constant**)



EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam





A. Emergency Depressurization is required if Torus water level rose and stabilized at 180" IAW the Primary Containment Control EOP.

B. Emergency Depressurization is required if Torus water temperature rose and stabilized at 175 °F IAW the Primary Containment Control EOP.

C. Rapidly depressurize the RPV with Turbine Bypass Valves in anticipation of Emergency Depressurization if Torus pressure rose and stabilized at 20 psig IAW the RPV Control - No ATWS EOP.

D. Rapidly depressurize the RPV with the Isolation Condensers in anticipation of Emergency Depressurization if Torus water level lowered and stabilized at 120" IAW the RPV Control - No ATWS EOP.

Answer: B

Answer Explanation:

QID: 09-1 NSRO10		
Question # / Answer	**10**	**Developer/Date: NTP 1/5/10**

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
295013 High Suppression Pool Temperature 2.1.23 - Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation					4.4
Level	SRO	Tier	1	Group	2
General References	EOP Users Guide				

Explanation	The plant was at power when an event occurred. Only 1 parameter is changed at a time with all others remain the same. At the RPV pressure of 700 psig, and Torus water level of 164", if Torus water temperature were to rise to 175 °F, then this would violate the Heat Capacity Limit Curve and an ED would be required. Answer B is correct. If Torus water level rose to 180", at 700 PSIG and a temperature of 155 °F, HCTL is not violated (temperature limit at 180" is 160 °F). Answer A is incorrect. At a Torus water level of 164" and Torus pressure rising to 20 psig, the Pressure Suppression Pressure is not violated and no ED would be required, thus anticipating ED is not permissible. Answer C is incorrect. At 700 psig, and a level of 120", the HCTL limit is 170 °F and the HCTL is not violated and thus ED is not required and cannot be anticipated. Also, since Torus water level is being maintained steady at > 110", ED is not required. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.845.0.0056 LO 200-10445	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPR**
	NUREG 1021 Appendix B: Solve a problem using a reference			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility conditions and selection of procedure			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

11 **ID: 09-1 NSRO11** **Points: 1.00**

The plant was shutdown with fuel shuffling underway when a fire occurred in the Boiler House. The fire heavily damaged both MCCs 1A24 and 1B24 and both are de-energized.

Which of the following states the impact on the refuel floor activities?

A. Core alterations are **not** impacted by the event and may continue unrestricted.

B. Core alterations may continue for **only** the next 7 days due to the loss of SGTS Fan A.

C. Core alterations may continue **only** for the next 7 days due to the loss of SGTS Fan B.

D. Core alterations shall cease immediately due to the loss of Secondary Containment integrity.

Answer: D

Answer Explanation:

QID: 09-1 NSRO11		
Question # / Answer	11	Developer/Date: NTP 1/6/10

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
261000 SGTS A2.07 - Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. electrical failure					2.8
Level	SRO	Tier	2	Group	1
General References	205 330	TS 1.14		TS 3.5.B	

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

Explanation	The plant is shutdown with fuel shuffling in-progress. This activity requires Secondary Containment integrity, which includes the Standby Gas Treatment System operable. SGTS Fan 1-8 is powered from MCC 1A24 and SGTS Fan 1-9 is powered from MCC 1B24. Refuel activities could continue for a limited time if only 1 fan were operable, but it is given that both MCCs 1A24 and 1B24 are de-energized, and are thus must be declared inoperable. Thus, there is no operable SGT Fan. TS definition 1.14 for Secondary Containment integrity includes SGTS operable. Procedure 205 also requires Secondary Containment integrity. With no SGTS, there is no Secondary Containment integrity and core alterations must cease immediately. Answer D is correct. Answers B & C would be correct for the loss of a single SGTS Fan but are incorrect for the loss of both. If the candidate does not realize the relationship between the MCCs and SGTS and Secondray Containment integrity, then answer A might seem correct. But, answer A is incorrect. 4/9/10: This question was modified following NRC feedback.		
References to be provided during exam:	**Reference Deleted**		
Learning Objective	2621.812.0.0003 LO 234-10451		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 2:DR**
	NUREG 1021 Appendix B: Describe or recognize relationships			
10CRF55 Content	**55.41**		**55.43**	**2**
	(SRO Only) Facility operating limitations in the Tech Specs and their bases			
Time to Complete: 1-2 minutes				

12 **ID: 09-1 NSRO12** **Points: 1.00**

The plant was starting up after an outage with reactor power monitored by the SRMs and has **not** been declared critical. The URO had just taken the shift and reported the following observations:

- SRM 21 shows a slow rise with **no** control rod selected
- SRM PERIOD CHANNEL 21 shows a slow rise (positive period getting shorter)
- All other SRMs indicate constant counts
- No annunciators have alarmed

Which of the following states the cause for the observations and the SRO direction to the URO?

	Cause	SRO Direction
A.	The applied voltage to the detector is lowering and the SRM is inoperable	Cease all control rod withdrawals due to inadequate operable SRMs
B.	The SRM 21 recorder is failing and the SRM 21 recorder is inoperable	Bypass SRM 21 and continue with control rod withdrawals
C.	The applied voltage to the detector is rising and the SRM is inoperable	Bypass SRM 21 and continue with control rod withdrawals
D.	A control rod is drifting outward from the core	Enter and execute ABN-6, Control Rod Malfunctions

Answer: C

Answer Explanation:

QID: 09-1 NSRO12		
Question # / Answer	**12**	**Developer/Date: NTP 1/6/10**

Knowledge and Ability Reference Information							
K&A						Importance Rating	
						RO	SRO
215004 Source Range Monitor **A2.01 - Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Power supply degraded**							2.9
Level	**SRO**	**Tier**	**2**	**Group**	**1**		
General References	**TS 3.2.B.5**		**GFES RAP-G4d**		**LP 2621.828.0.0029**		
Explanation	The plant is starting up with reactor power on the SRMs when an observation is reported that SRM 21 is rising and the associated period meter is also rising. Both the SRM indication and period meter are fed from the SRM drawer. The SRMs operate in the Proportional region of the Gas-Filled Detector Characteristic Curve. As applied voltage to the detector rises, the counts also rises. Since no control rods are currently selected and being withdrawn, the SRMs will read a constant value. Since SRM 21 is displaying aberrant behavior, it shall be declared inoperable and bypassed as directed by the RAP when it alarms from high counts. IAW TS 3.2.B.5, only 3 operable SRMs are required, and thus the SRO can allow control rod withdrawals to continue. Answer C is correct. Answer A is incorrect since a lower detector voltage will result in fewer counts and there is no need to cease control rod withdrawals. Answer A is incorrect. Since the SRM drawer feeds the SRM indicator and the period meter, if just the SRM indicator was failing, there would be no change in the period meter. Answer B is incorrect. The question stem says that no annunciators have alarmed. It is true that a drifting outward control rod would give the indications observed in the question stem, the rod drift annunciator would be in alarm. Therefore, it can't be due to a drifting control rod. Answer D is incorrect.						

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

References to be provided during exam:	**None**	

Learning Objective	2621.812.0.0003 LO 234-10451

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility conditions and selection of appropriate procedure			
Time to Complete: 1-2 minutes				

13 **ID: 09-1 NSRO13** **Points: 1.00**

The plant was at rated power when an event occurred. Present plant conditions include the following:

- Offsite power has been lost
- EDG 1 has experienced an overspeed condition
- The RPV has been depressurized to 0 psig
- RPV water level is -10" and steady
- Drywell pressure is 21 psig
- Drywell temperature is 268 °F
- Torus water temperature is 104 °F
- Fire Water is injecting into the RPV via Core Spray System 1
- Core Spray System 2 s injecting into the RPV at 2000 GPM

Which of the following is the SRO's **next** direction?

A. Lineup CST to the Core Spray System.

B. Confirm SLC Pump B is injecting into the RPV.

C. Enter the Primary Containment Flooding Procedure.

D. Initiate Drywell Sprays with Containment Spray System A.

Answer: B

Answer Explanation:

QID: 09-1 NSRO13		
Question # / Answer	**13**	**Developer/Date: NTP 1/6/10**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
211000 SLC **2.4.4 - Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.**						**4.7**
Level	**SRO**	**Tier**	**2**	**Group**	**1**	

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

General References			
Explanation	The plant was at rated power when an event occurred, with several plant conditions given: Drywell temperature and pressure are high, Torus water temperature is high, and RPV water level is low. Offsite power is gone which removes and Feedwater/Condensate flow. EDG 1 & 2 fast started from the loss of offsite power, but EDG 1 has since tripped on overspeed. Most faults are bypassed on a EDG fast start, except overspeed. Thus Bus 1C and downstream busses are de-energized. IAW the RPV Control - No ATWS EOP, when RPV water level lowered to 0", the RPV was emergency depressurized. With RPV water level at 10", the nuclear fuel is exposed and adequate core cooling becomes the priority. Under the conditions given, with RPV water level this low, no FW/Condensate and Core Spray injecting at much less than design values, the EOP again directs SLC injection. SLC Pump 2 is powered from USS 1B2 (from EDG 2) and is available. Answer C is correct. Lining-up the CST to Core Spray is a potential path in the RPV Control - No ATWS EOP, but Core Spray B (System 2) should be and is injecting. CST could be lined up to those Core Spray Pumps not already injecting, but Fire Water is already injecting through the other Core Spray System. Fire water is a higher pressure, higher capacity that the CST flow. Thus answer A is incorrect. Under the given conditions, if RPV water level cannot be maintained > -20", then Primary Containment Flooding would be a correct answer. But with RPV water level at -10" and steady, answer C is incorrect. Spraying the Drywell is allowed under the conditions given, but with no offsite power and EDG 1 tripped, there is no power to Containment Spray A. Answer D is incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.845.0.0052 LO 200-10445		

Question Source (New, Modified, Bank)	New

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SPK
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	55.41		55.43	5
	(SRO Only) Assessment of facility conditions and selection of appropriate procedure			
Time to Complete: 1-2 minutes				

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

14 **ID: 09-1 NSRO14** **Points: 1.00**

The plant was at rated power when an event occurred. The Operator reports the following observations:

- BUS 1B CNTRL DC LOST has alarmed
- BUS 1D CNTRL DC LOST has alarmed
- **All** Isolation Condenser A valves on Panel 1F/2F indicate green light on
- Annunciator DC-E PWR XFER has **not** alarmed
- Annunciator DC-D PWR XFER has **not** alarmed

Which of the following shall the SRO direct?

Note:

ABN-53 is DC A and Panel Failures
ABN-54 is DC B and Panel/MCC Failures
ABN-55 is DC C and Panel/MCC Failures

A. IAW ABN-54, direct an Operator to manually align DC-1 transfer switch to DC-A.

B. IAW ABN-53, direct an Operator to manually align DC-E transfer switch to DC-B.

C. IAW ABN-54, direct an Operator to manually align DC-D transfer switch to DC-A.

D. IAW ABN-55, direct an Operator to manually operate supply and load breakers at DC-2 as required.

Answer: C

Answer Explanation:

QID: 09-1 NSRO14		
Question # / Answer	**14**	**Developer/Date: NTP 1/6/10**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

263000 DC Electrical Distribution 2.1.30 - Conduct of Operations: Ability to locate and operate components, including local controls.						4.0
Level	SRO	Tier	2	Group	1	

General References	ABN-54	ABN-53	ABN-55
Explanation	The plant was at rated power when DC control power was lost to Bus 1B & 1D. This DC power comes from 125 VDC Bus DC-B. Thus, there is a loss of DC-B and ABN-54 applies. The provided information also states that DC-D & DC-E did not transfer to their alternate DC supply. Of these, only DC-D is fed from DC-B. The question stem also states that the valve positions for Isolation Condenser A indicate their normal positions. Two of the valves are powered by DC-1, which is fed by DC-B. Since the indications do show valve positions, then DC-1 has transferred to its alternate DC supply (DC-A). Thus, DC-B has been lost and DC-D did not auto transfer. IAW ABN-54, manually performing the transfer of DC-D is correct. Answer C is correct. ABN-54 does direct manually transferring the power supply for DC-1 if it didn't auto transfer. Bus as shown, it did auto transfer. Answer A is incorrect. DC-E is normally powered from DC-A and the alternate supply is DC-B. But since DC-A has not lost power, performing actions IAW ABN-53 is not appropriate. Answer B is incorrect. The action in answer D is correct IAW ABN-55, but this ABN will not be entered under the given conditions since DC-C has not lost power. Answer D is incorrect.		
References to be provided during exam:	None		
Learning Objective	2621.828.0.0012 LO 263-10445		

Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 3:SKP
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

10CRF55 Content	55.41		55.43	5
	(SRO Only) Assessment of facility conditions and selection of appropriate procedure			
Time to Complete: 1-2 minutes				

15 **ID: 09-1 NSRO15** **Points: 1.00**

The plant was at rated power when a major unisolable TBCCW leak occurred. **All** IMMEDIATE OPERATOR ACTIONs of ABN-20, TBCCW FAILURE RESPONSE, have been performed. Present plant conditions include the following:

- RPV water level is 168" and steady
- RPV pressure is 900 psig and steady
- **All** control rods in core quadrant 1 indicate position 04 and **all** other control rods indicate full-in
- Primary Containment parameters indicate normal

The Operator reports the following observation:

- **All** operating Feedwater and Condensate Pump temperatures indicate in excess of 250 °F

Which of the following will the SRO direct **next**?

A. Swap operating Feedwater Pumps IAW procedure 317 due to high bearing temperatures.

B. Terminate and prevent Feedwater injection IAW the Support Procedure, due to the ATWS.

C. Trip **only** the operating Feedwater Pumps IAW ABN-1, Reactor Scram, due to RPV high water level.

D. Trip **both** the operating Feedwater **and** Condensate Pumps IAW the ABN, due to high bearing temperatures.

Answer: D

Answer Explanation:

QID: 09-1 NSRO15		
Question # / Answer	15	Developer/Date: NTP 1/7/10

Knowledge and Ability Reference Information		
K&A	Importance Rating	
	RO	SRO

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

400000 Component Cooling Water 2.4.16 - Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.						**4.4**
Level	**SRO**	**Tier**	**2**	**Group**	**1**	
General References	**RAP-H8j**		**EOP Users Guide**		**ABN-20**	

Explanation	The plant was at rated power when a major unisolable TBCCW leak occurred. In this condition, ABN-20 requires a manual scram and trip of all recirculation pumps. Conditions show that an entire quadrant of control rods inserted only to position 04. TBCCW supplies cooling to the Feedwater and Condensate Pumps. It is expected that the temperatures of the operating Feedwater & Condensate Pumps would rise. The conditions show normal parameters following a scram for RPV water level, pressure and Primary Containment, and with RPV water level within the EOP prescribed band and no RPV leaks in the primary Containment. On a normal scram, it is expected that the RPV Control - No ATWS EOP will be entered on RPV water level. And in this case, with many control rods at position 04, the reactor can still be determined to be shutdown and the No ATWS EOP is the correct EOP. In the water level leg of the EOP, it directs controlling water level using Feedwater/Condensate, CRD, and/or Core Spray. Each of these systems is bulletized which means that no system takes precedence over any other. 1 Feedwater Pump and all 3 Condensate Pumps will be running post-scram. The ABN-20 says to secure all pumps at the given temperatures. Since there is no RPV leak, RPV water level is steady high in the band, and both CRD pumps remain available, then stopping all pumps IAW the ABN is correct. Answer D is correct. Swapping Feedwater Pumps is plausible since 2 Feedwater Pumps would already have been shutdown. But, since all running condensate pumps would also be overheated, they would require to be shutdown as well. Running a Feedwater Pump with no condensate pump is not allowed. Thus answer A is incorrect. In the ATWS EOP, it is appropriate to terminate and prevent Feedwater injection when power is >2%. Although no power level is given, the candidate may think that entry into the ATWS EOP is correct and termination of Feedwater could be correct IAW Support Procedure 17. Since the No ATWS EOP allows control rods at position 04, then the ATWS EOP is incorrect and no termination/prevention of Feedwater IAW the Support procedures applies. Answer B is incorrect. ABN-1 provides direction for RPV water level control. When water level cannot be restored and maintained

	below 170", it directs to trip all Feedwater Pumps. But with water level at 168" and steady, this is not appropriate. Answer C is incorrect.	
References to be provided during exam:	**None**	
Learning Objective	2621.828.0.0017 LO 260-10445	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility conditions and selection of appropriate procedures			
Time to Complete: 1-2 minutes				

16 **ID: 09-1 NSRO16** **Points: 1.00**

The plant was at rated power with Offgas Radiation Monitor RN12A failed downscale due to an electrical failure.

An event then occurred that resulted in the following annunciators alarming over the next 3 minutes:

- RWCU HELB I **and** RWCU HELB II
- RADIATION MONITORS PROCESS - OFFGAS HI **and** OFFGAS HI HI
- RADIATION MONITORS AREA - AREA MON HI

The Operator reports the following observations:

- All Reactor Water Cleanup System isolation valves indicate green light on
- Offgas Radiation Monitor RN12B indicates 1100 mr/hr and **rising** slowly
- STACK EFFLUENT LRM CH#1 and CH#2 are **rising** slowly
- ARM CLEANUP SYS PUMP AREA (C-1) indicates 40 mr/hr and **lowering**

Which of the following states the impact on the Offgas System and the next SRO direction?

	Offgas Impact	Required Action
A.	Offgas will automatically isolate	Reduce reactor power IAW ABN-ABN-26, High Main Steam/Offgas/Stack Effluent Activity
B.	Offgas will automatically isolate	Scram the reactor IAW the Secondary Containment Control EOP
C.	Offgas will **not** automatically isolate	Scram the reactor IAW the Radioactivity Release Control EOP
D.	Offgas will **not** automatically isolate	Shutdown the reactor IAW the Secondary Containment Control EOP

Answer: A

Answer Explanation:

QID: 09-1 NSRO16		
Question # / Answer	16	Developer/Date: NTP 1/7/10

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
272000 Radiation Monitoring A2.03 - Ability to predict the impacts of the following on the RADIATION MONITORING SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. electrical failure					3.1
Level	SRO	Tier	2	Group	2
General References	ABN-26	EOP Users Guide	EP-AA-1010		

<table>
<tr><td>Explanation</td><td colspan="2">The plant was at rated power with an offgas radiation monitor failed downscale due to a power failure. An event then occurs which shows a leak in the RWCU System, high radiation in the offgas system, an area radiation monitor in alarm, and increased radiation in the stack. Several procedures could be entered: the RAP for the alarms, ABN-26, Secondary Containment Control EOP, and possibly the Radioactive release Control EOP.
The Operator observations show that the RWCU System isolated, as designed; 1 offgas radiation monitor is above the high setpoint, and the RWCU ARM radiation indication is lowering (confirming that the RWCU leak has been stopped).
The logic for the offgas radiation monitors for isolating the offgas system is both hi-hi, or one downscale and 1 hi-hi. Thus, with the information provided, 1 radiation monitor is hi-hi and 1 is downscale. Therefore, the offgas isolation will occur (after a 15 minute timer times out).
For the conditions provided, ABN-26 directs lowering reactor power in an attempt to clear the alarm to prevent the offgas system isolation. Answer A is correct.
The Secondary Containment Control EOP directs a scram prior to exceeding a max safe value, provided that a primary system is discharging into the Secondary Containment. From the question, it shows that a primary system was discharging into the Secondary Containment but is no longer, and radiation levels are lowering. Thus a scram is not the appropriate action. Answer B is incorrect.
Indications show that an offsite radiological release is in progress, but it does not rise to the point of the entry condition for the Radioactive Release Control EOP. If it had, then a scram would be appropriate. Also, offgas will isolate. Answer C is incorrect.
A reactor shutdown is appropriate in the Secondary Containment Control EOP if a non-primary system were discharging into the Secondary Containment and radiation levels were rising. Also, offgas will isolate. Answer D is incorrect.</td></tr>
<tr><td>References to be provided during exam:</td><td>None</td><td></td></tr>
<tr><td>Learning Objective</td><td colspan="2">2621.828.0.0004 LO 193</td></tr>
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Question Source (New, Modified, Bank)			New	
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X 2:DR
	NUREG 1021 Appendix B: Describe or recognize relationship			
10CRF55 Content	55.41		55.43	5
	(SRO Only)			
Time to Complete: 1-2 minutes				

17 | **ID: 09-1 NSRO17** | **Points: 1.00**

The plant was starting up after a forced outage. Present plant conditions include the following:

- The REACTOR MODE SELECTOR switch is in STARTUP
- RPV pressure indicates 0 psig
- RECIRC PUMP SUCTION TEMPS indicates 60 °F subcooled
- The very **first** control rod in the pull sheet has been withdrawn to position 48

An event then occurred resulting in the following:

- An apparent electrical malfunction caused TIP #1 to drive to the CORE TOP and remain
- Efforts to withdraw the TIP from the Control Room were unsuccessful
- Control rod withdrawals have been halted

IAW TS 3.5, Containment, which of the following actions is correct?

A. The TIP ball valve must be closed within 4 hours or activate the shear valve.

B. The TIP ball valve must be closed within 48 hours or activate the shear valve.

C. No actions are required since Primary Containment Integrity is not currently required.

D. The TIP ball valve **or** shear valve must be closed within 48 hours or insert all control rods.

Answer: C

Answer Explanation:

QID: 09-1 NSRO17		
Question # / Answer	**17**	**Developer/Date: NTP 1/9/10**

Knowledge and Ability Reference Information		
K&A	**Importance Rating**	
	RO	**SRO**

215001 Traversing In-core Probe **2.2.22 - Equipment Control: Knowledge of limiting conditions for operations and safety limits**							**4.7**
Level	**SRO**		**Tier**	**2**	**Group**	**2**	
General References	**TS 3.5**						

Explanation	The plant is starting after a refuel outage with 1 control rod withdrawn. Shutdown margin proves that the reactor cannot go critical with a single control rod withdrawn. The stem also shows that RPV pressure is 0 psig and is 60 °F subcooled. This means that RPV coolant temperature is 212-60 = 152 °F. An event then occurs which results in TIP #1 driving in to the Core Top location, and it cannot be retracted. TS 3.5 provides the following: PRIMARY CONTAINMENTINTEGRITY shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt or during REACTOR VESSEL PRESSURE TESTING. a. With one or more of the automatic containment isolation valves inoperable: (1) Maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours (48 hours for the traversing in-core probe system) either; (a) Restore the inoperable valve(s) to OPERABLE status or (b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position, or (c) Isolate each affected penetration by use of at least one closed manual valve or blind flange. (2) If Specification 3.5.A.3 or the provisions of Specifications 3.5.A.3.a.(1)(a), (b) or (c) can not be met, the reactor shall be PLACED IN the COLD SHUTDOWN CONDITION within 24 hours. Under the given conditions, Primary Containment Integrity is not required and thus there are no TS actions required. Answer C is correct. Had Primary Containment Integrity been required, then LCOs for an inoperable TIP would have applied. The other answers allow some time to recover primary Containment but are incorrect. 4/9/10: This question was modified following NRC feedback. Changed 'refuel outage' to 'forced outage' in the question stem.	
References to be provided during exam:	**Reference Deleted**	
Learning Objective	2624.828.0.0032 LO 422	

<table>
<tr><td colspan="3">Question Source (New, Modified, Bank)</td><td colspan="2">New</td></tr>
<tr><td rowspan="2">Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X
3:SPR</td></tr>
<tr><td colspan="4">NUREG 1021 Appendix B: Solve a problem using a reference</td></tr>
<tr><td rowspan="2">10CRF55 Content</td><td>55.41</td><td></td><td>55.43</td><td>2</td></tr>
<tr><td colspan="4">(SRO Only) Facility operating limitations in the Tech Specs and their basis</td></tr>
<tr><td colspan="5">Time to Complete: 1-2 minutes</td></tr>
</table>

18 **ID: 09-1 NSRO18** **Points: 1.00**

The plant was at 15% power on a startup. Present plant conditions include the following:

- Feedwater Pump C and Condensate Pumps A & C in service
- CRD Pump NC08A is in service

An event then occurred resulting in the loss of DC Bus B.

While investigating the electrical problem, A LOCA in the Primary Containment occurred. Present plant conditions include the following:

- RPV water level is 125" and lowering slowly
- RPV pressure is 825 psig and lowering slowly
- Drywell temperature is 248 °F and rising
- Drywell pressure is 17 psig and rising slowly
- Torus water temperature is 87 °F and steady

Which of the following is the **next** SRO direction?

A. Start CRD Pump NC08B and maintain RPV water level 138" - 175" IAW the RPV Control - No ATWS EOP.

B. Start Feedwater Pump A and maintain RPV water level 138" - 175" IAW the RPV Control - No ATWS EOP.

C. Emergency Depressurize the RPV IAW the Drywell Temperature leg of the Primary Containment Control EOP.

D. Initiate Drywell Sprays using Containment Spray System 2 IAW the Drywell Pressure leg of the Primary Containment Control EOP.

Answer: B

Answer Explanation:

QID: 09-1 NSRO18		
Question # / Answer	**18**	**Developer/Date: NTP 1/8/10**

Knowledge and Ability Reference Information

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

K&A					Importance Rating	
					RO	SRO
259001 Reactor Feedwater A2.08 - Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of DC electrical power						2.6
Level	SRO	Tier	2	Group	2	
General References	BR 3028		RPV Control - No ATWS EOP		317	

Explanation	The plant was at 15% power on a startup when DC B was lost. Then, a LOCA in the Primary Containment occurred. Indications provided show that RPV water level is low and Drywell parameters are elevated. A loss of DC B results in the loss of control power to Bus 1B, 1D, USS 1B2 and 1B3. With RPV water level low, the RPV Control EOP directs RPV water level 138-175" with Support procedure 2 (Feedwater) and SP-3 (CRD). With the loss of DC B, there is no power to close either Feedwater Pump B or C breaker from the control room. Feedwater Pump A gets DC control power from DC C, which is available. Answer B is correct. SP-3 for CRD does say to start additional CRD pumps to maintain RPV water level. But with no control power, the Panel Operator is unable to start CRD Pump NC08B (DC supply is from DC B). Answer A is incorrect. The Drywell temperature leg of the Primary Containment Control EOP directs ED if Drywell temperature cannot be maintained less than 281 °F. The DC loss prevents starting any Containment Spray System 2 pumps from the Control Room, but System 1 pumps are available and can be started from the control room. This avenue should be attempted first to see if Drywell temperature can be maintained less than the limit to prevent the need to ED as given in answer D. Answer C is incorrect. As discussed for answer C, Containment Spray System 2 pumps cannot be started from the control room to lower Drywell temperature, but System 1 pumps can be. These Containment Spray Pumps shall be attempted to lower Drywell temperature and pressure before it is determined that Drywell temperature cannot be maintained below 281 °F, and the direction to Emergency Depressurize is required. Answer D is incorrect.

References to be provided during exam:	**None**	
Learning Objective	2621.845.0.0052 LO 3055	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**

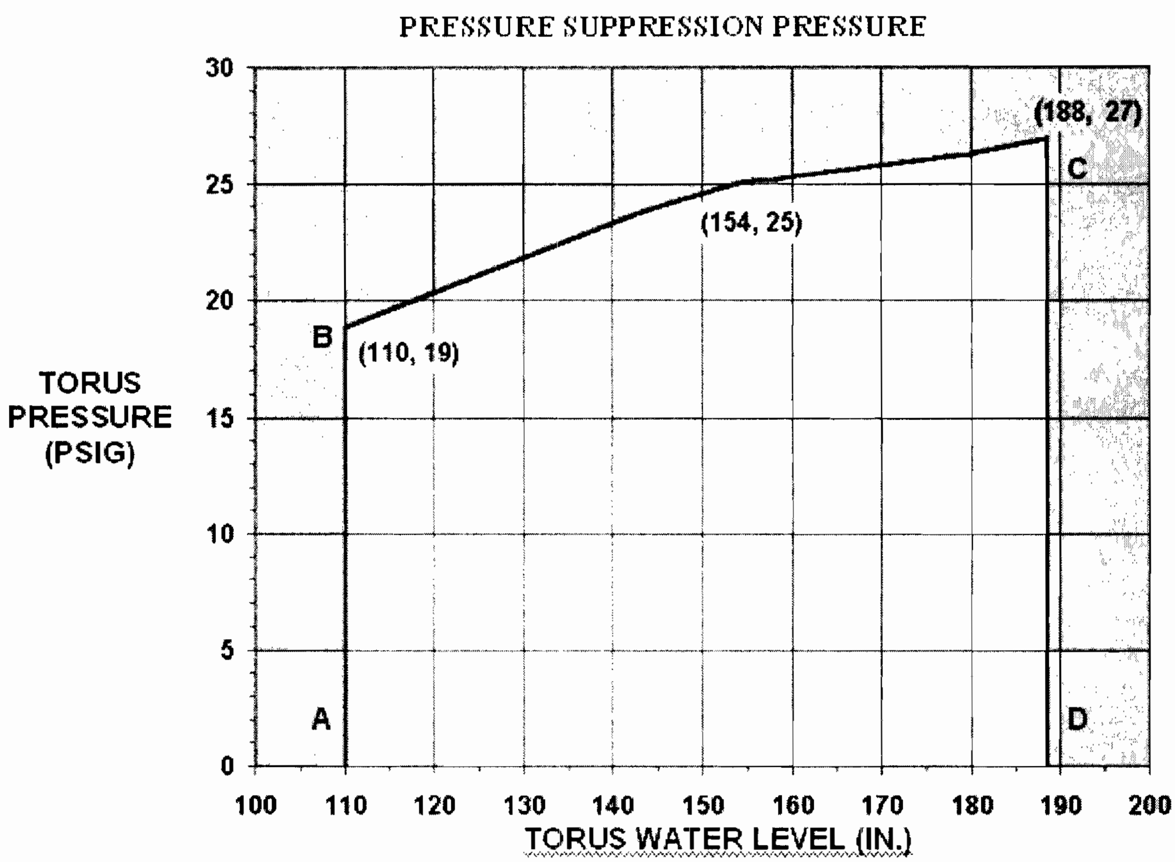
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning				
	10CRF55 Content	55.41		55.43	5
		(SRO Only) Assessment of facility conditions and selection of appropriate procedure			
	Time to Complete: 1-2 minutes				

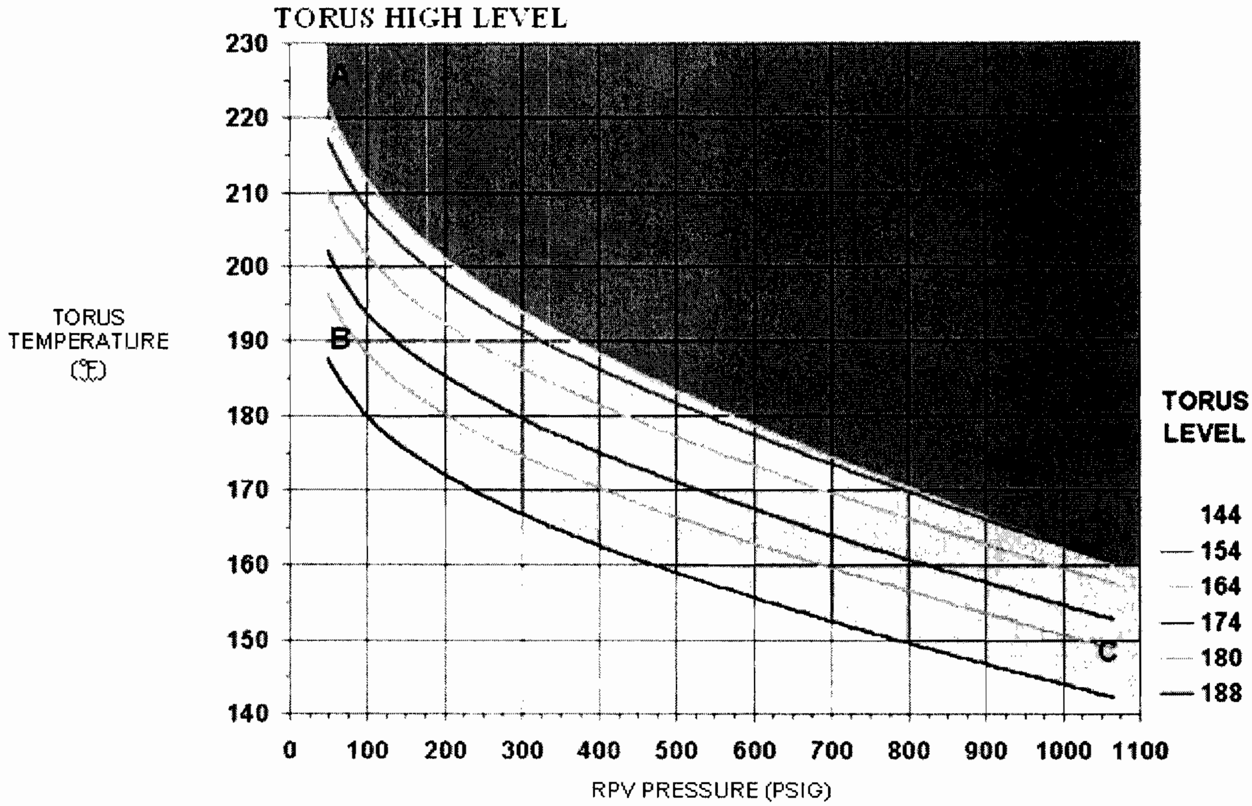
19 **ID: 09-1 NSRO19** **Points: 1.00**

The reactor was at rated power when a LOCA occurred. Present plant conditions include the following:

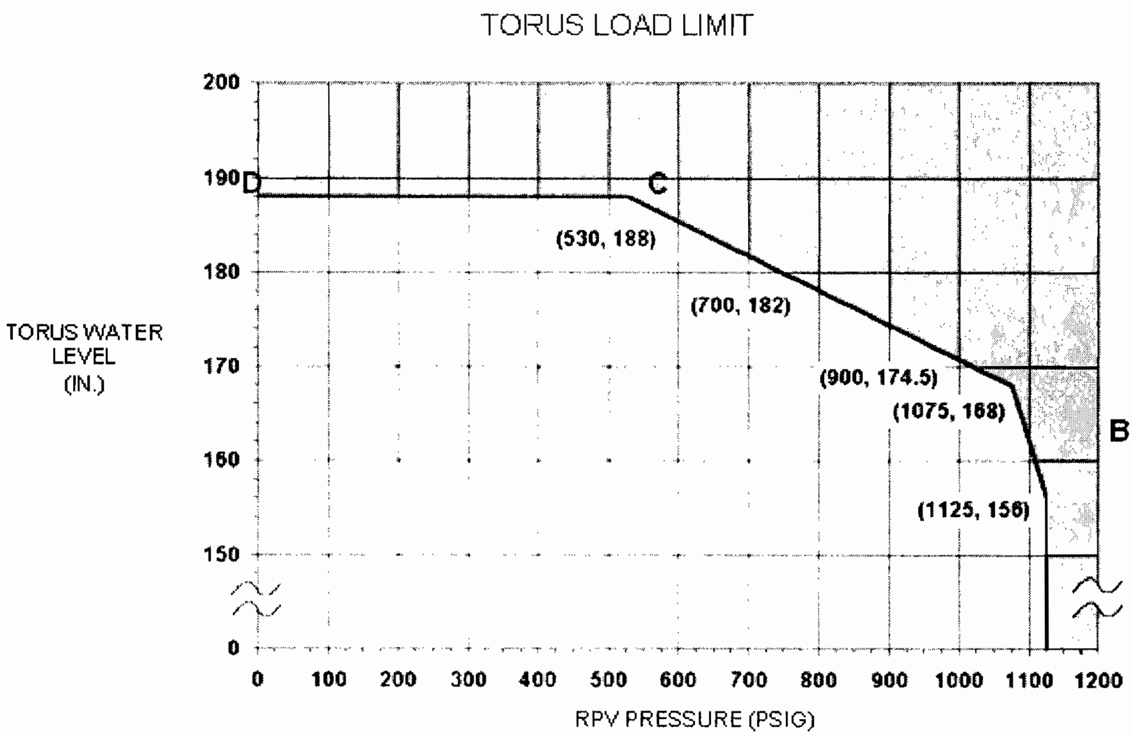
- All control rods inserted
- RPV pressure is 45 psig
- RPV water level is 60" and rising
- 1 Condensate Pump is injecting
- Core Spray A and B are injecting
- Torus water level is 184" and rising
- Torus pressure is 22 psig and rising slowly
- Containment Sprays are inoperable

Which of the following shall the SRO direct **next**?





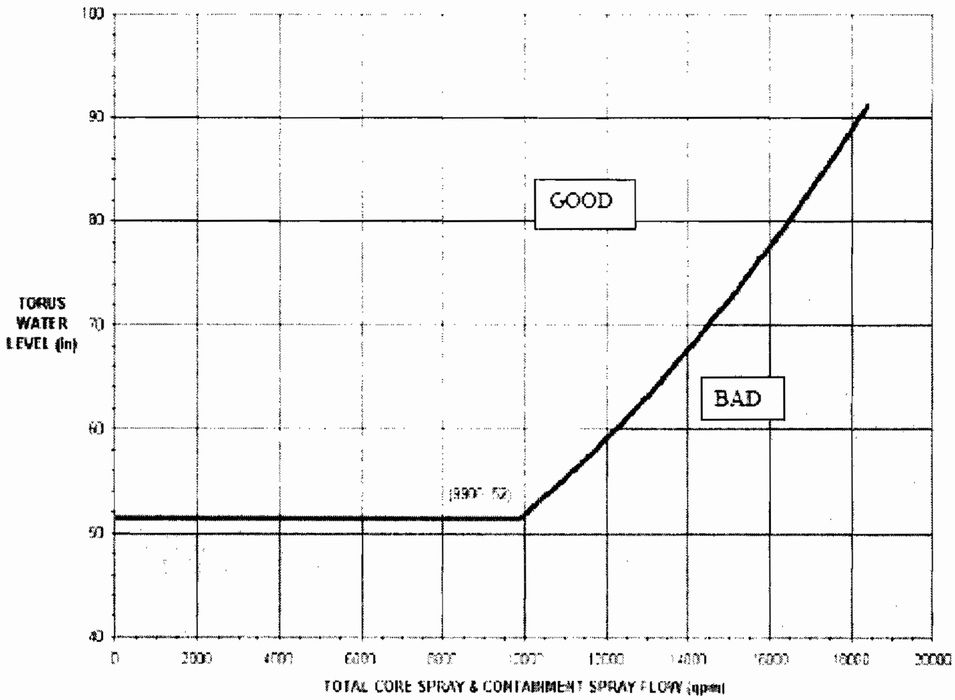
TORUS HIGH LEVEL
230
220
210
200
190
180
170
160
150
140
TORUS TEMPERATURE (°F)
A
B
C
0 100 200 300 400 500 600 700 800 900 1000 1100
RPV PRESSURE (PSIG)
TORUS LEVEL
144
154
164
174
180
188



TORUS LOAD LIMIT
200
190
180
170
160
150
0
TORUS WATER LEVEL (IN.)
D
C
B
(530, 188)
(700, 182)
(900, 174.5)
(1075, 168)
(1125, 156)
0 100 200 300 400 500 600 700 800 900 1000 1100 1200
RPV PRESSURE (PSIG)

FIGURE A

CORE SPRAY VORTEX LIMIT



A. Due to Vortex limits concerns, terminate Core Spray IAW Support Procedure 4, Operation of the Core Spray System.

B. Due to Heat Capacity Temperature Limit concerns, Emergency Depressurize using the EMRVs IAW the Emergency Depressurization - No ATWS EOP.

C. Due to Torus Load Limit concerns, terminate RPV injection with Condensate and continue injection with Core Spray IAW the Primary Containment Control EOP.

D. Due to Pressure Suppression Pressure limit concerns, anticipate Emergency Depressurization and rapidly depressurize with the Turbine Bypass valves IAW the RPV Control - No ATWS EOP.

Answer: C

Answer Explanation:

QID: 09-1 NSRO19		
Question # / Answer	**19**	**Developer/Date: NTP 1/9/10**

Knowledge and Ability Reference Information							
K&A						Importance Rating	
						RO	SRO
Conduct of Operations **2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.**							4.4
Level	**SRO**	**Tier**	**3**	**Group**			
General References	**EOP Users Guide**		**LP 2621.828.0.0005**				

Explanation	The plant was at rated power when a LOCA occurred. Present plant conditions include low RPV water level with 1 condensate pump and Core Spray injecting; Torus high temperature/pressure and high water level. IAW the Primary Containment Control EOP, the Torus Load Level is being approached. At the current RPV pressure, lowering RPV pressure adds no additional margin to TLL. Thus the EOP directs stopping injection into the RPV from sources external to the Primary Containment as long as adequate core cooling is maintained. Condensate is outside the Primary Containment and adequate core cooling is currently assured with water level at 60" and rising. Answer C is correct. Core Spray should be secured IAW Support Procedure 4 if vortex limits are exceeded. Even though no Core Spray flow was provided, it can be seen that the limit is not exceeded. Thus answer A is incorrect. If HCTL is exceeded, the EOP directs ED. But given an RPV pressure of only 45 psig, the EMRVs will not open. Answer B is incorrect. Because an ED limit is being approached, anticipation of ED and rapidly depressurizing the RPV is allowed in the EOPs. Bus since RPV water level is 60", and there is no allowed bypass for MSIV closure on low-low RPV water level (as in the ATWS EOP), the MSIVs have closed on low-low water level are the turbine bypass valves are not available. Answer D is incorrect.

References to be provided during exam:	**None**	

Learning Objective	2621.845.0.0056 LO 200-10445

Question Source (New, Modified, Bank)				Modified
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	x 3:SPK
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning			
10CRF55 Content	55.41		55.43	5
	(SRO Only) Assessment of facility conditions and selection of procedure			
Time to Complete: 1-2 minutes				

20 **ID: 09-1 NSRO20** **Points: 1.00**

The plant was at rated power. You are reviewing scheduled work for the following day. You note that the removal of CRD Pump NC08A from service for a scheduled PM places the Plant Status Risk Color in Red.

IAW WC-OC-101-1001, On-Line Risk Management and Assessment, which of the following is correct regarding the scheduled removal of CRD Pump NC08A from service?

A. Perform the activity around the clock.

B. Pre-stage **all** required parts and materials.

C. Identify **all** associated protected equipment.

D. CRD Pump NC08A can **not** be removed from service as scheduled.

Answer: D

Answer Explanation:

QID: 09-1 NSRO20		
Question # / Answer	**20**	**Developer/Date: NTP 1/10/10**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
Equipment Control **2.2.17 - Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, coordination with the transmission system operator.**						**3.8**
Level	**SRO**	**Tier**	**3**	**Group**		
General References	**WC-OC-101-1001**					

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

Explanation	The plant is at rated power when review of work activities shows that removal of a CRD Pump from service will change the plant status risk color to red. IAW the reference, a red risk condition is considered unacceptable and shall not be entered intentionally based on planned work activities. Therefore, removal of the CRD Pump shall not be allowed as planned. Answer D is correct. All other answers are conditions required for other risk colors (yellow and orange) and are plausible but incorrect.	
References to be provided during exam:	**None**	
Learning Objective		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility condition and selection of appropriate procedure			
Time to Complete: 1-2 minutes				

21 **ID: 09-1 NSRO21** **Points: 1.00**

The plant was shutdown for an outage.

As the SRO, you are performing a pre-job brief with an EO assigned to perform an in-plant task. The EO (a Declared Pregnant Woman), with a current annual exposure of 200 mrem, is to perform a job in the Drywell. Radiological conditions in the work area are 125 mrem/hr with an expected completion time of 3 hours.

IAW RP-AA-203, Exposure Control and Authorization, which of the following is correct?

A. The worker is **not** allowed to perform the job.

B. The worker is allowed to perform the job with **no** does extension required.

C. The worker is allowed to perform the job if a Dose Control Level Extension Form is first approved by the Station Manager.

D. The worker is allowed to perform the job if a Dose Control Level Extension Form is first approved by the Site Vice President.

Answer: A

Answer Explanation:

QID: 09-1 NSRO21		
Question # / Answer	**21**	**Developer/Date: NTP 1/10/10**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
Radiation Control **2.3.4 - Knowledge of radiation exposure limits under normal or emergency conditions.**						**3.7**
Level	**SRO**	**Tier**	**3**	**Group**		
General References	**RP-AA-203**					

Explanation	The plant is shutdown and a declared pregnant woman is selected to perform work in a radiological area. The work area is 125 mrem/hr and the job will take 3 hours, for a total exposure of 375 mrem. IAW the reference, the NRC limit for a declared pregnant woman is 500 mrem for the entire gestation period. Thus, the woman's total exposure would be 575 mrem, which is above the NRC limit. Thus, the woman cannot perform the job. Answer A is correct. If the candidate (incorrectly) applies the administrative dose control level to this worker (which is 2000 mrem), then the worker could perform the job with no dose extension required. Answer B is incorrect. To extend a workers exposure above the administrative dose level, a dose level extension form would be required. The approval signature required depends on how close to the NRC level the extension is for. The Station Manager and Site Vice President could be responsible as the approver for the extension form.

References to be provided during exam:	**None**	
Learning Objective		

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**		**55.43**	**4**
	(SRO Only) Radiation hazards that may arise during normal and abnormal situations			
Time to Complete: 1-2 minutes				

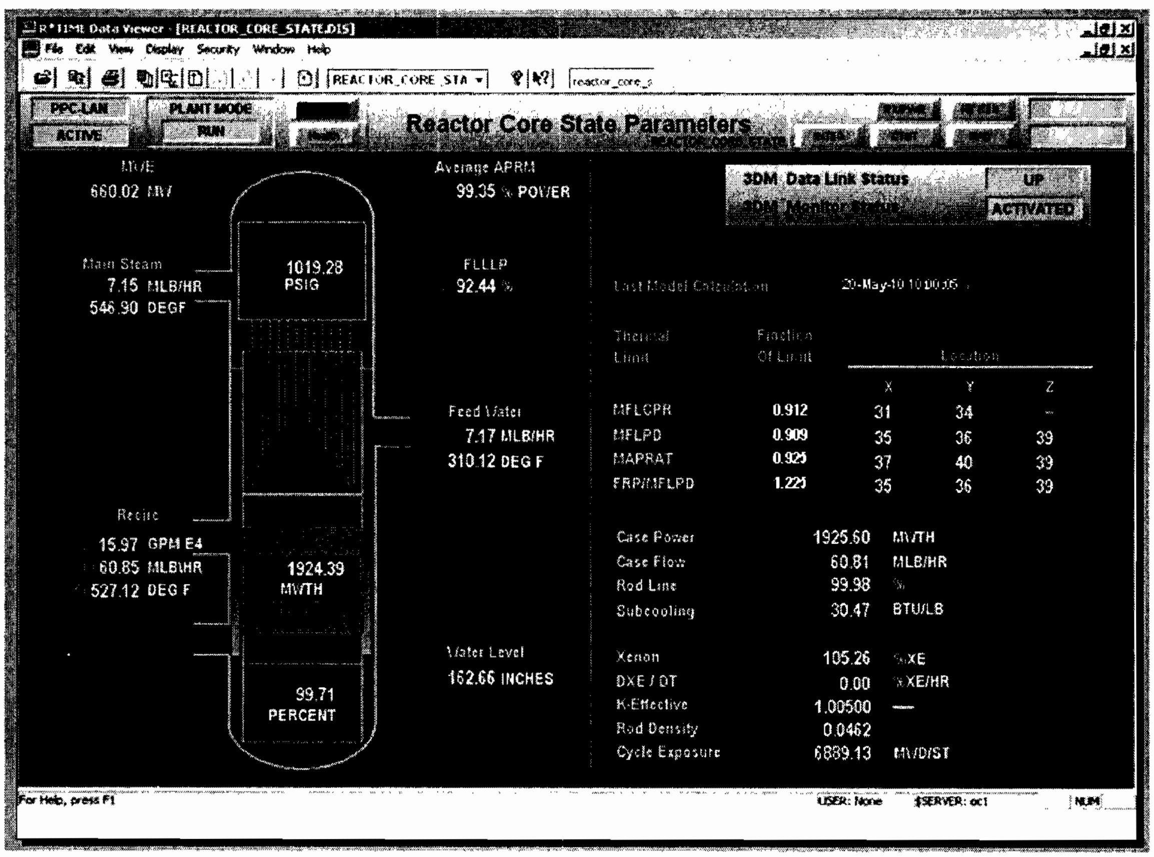
22 **ID: 09-1 NSRO22** **Points: 1.00**

The plant was at rated power when the following annunciator alarmed:

- CORE SPRAY 1 - SPARGER 1 DP HI

The EO reported that the local indication showed a reading of 2 PSID on Instrument Rack RK04.

IAW the RAP, which of the following states the impact on the Core Spray System and what action shall the SRO direct?



Core Spray Impact **SRO Direction**

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam

A.	Because of a Core Spray line break in the core region, consider Core Spray Loops A & C inoperable	Reduce reactor power to lower MAPRAT $\leq$ 0.9
B.	Because of a Core Spray line break in the annulus region, consider Core Spray Loops A & C inoperable	Reduce reactor power to lower MAPRAT $\leq$ 0.9
C.	Because of a Core Spray line break in the annulus region, consider Core Spray Loops A & B inoperable	Reduce reactor power to lower MFLPD $\leq$ 0.9
D.	Because of a Core Spray line break in the core region, consider Core Spray Loops A & B inoperable	Reduce reactor power to lower MFLPD $\leq$ 0.9

Answer: B

Answer Explanation:

QID: 09-1 NSRO22		
Question # / Answer	**1**	**Developer/Date: NTP**

Knowledge and Ability Reference Information						
K&A					**Importance Rating**	
					RO	**SRO**
Emergency Procedures/Plan 2.4.50 - Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.						**4.0**
Level	**SRO**	**Tier**	**3**	**Group**		
General References	**RAP-B5e**		**201 GE 885D781**		**TS 3.4.A**	

Explanation	The plant was at rated power when indications of a core spray line break occurred. This break impacts Core Spray loops A & C. The break is postulated to be in the Core Spray line in the annulus region. IAW the RAP, MAPRAT is to be lowered to 0.9 or below. The same requirement is provided in TS 3.4.A.3. Answer B is correct. If the candidate confuses which Core Spray loops are in System 1 (Containment Spray System 1 includes A & B loops), or confuses the location of the break, or confuses which thermal limit is affected, the other answers are plausible but incorrect. 4/9/10: This question was modified following NRC feedback.	
References to be provided during exam:	**Reference Deleted**	
Learning Objective	2621.828.0.0010 LO 209-10451	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**		**55.43**	**5**
	(SRO Only) Assessment of facility conditions and selection of appropriate procedure			
Time to Complete: 1-2 minutes				

23 **ID: 09-1 NSRO23** **Points: 1.00**

The plant was at power when an event occurred resulting in an offsite radiological release.

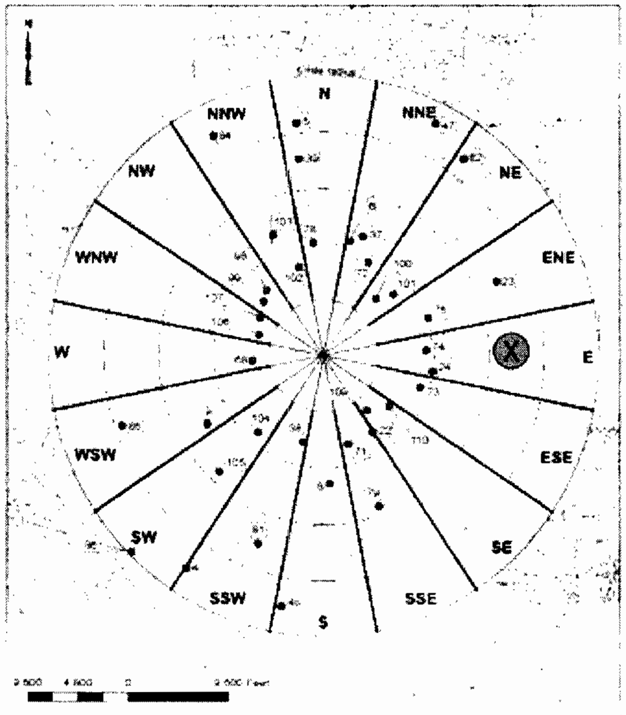
IAW Emergency Procedures, which of the following lists the **smallest** release and the wind direction **as indicated on the PPC**, which would require the Shift Emergency Director to **recommend evacuation** of Point X on the attached map?

The EP-AA-1010, Radiological Emergency Plan for Oyster Creek Station, Radiological Effluents thresholds is shown below. (Assume there are no offsite impediments to evacuation.)

Table R1 – Effluent Monitor Thresholds				
	GE	SAE	Alert	UE
Main Stack RAGEMS	4.0 E+01 μCi/cc HRM OR 1.6 E-08 amps HRM	4.0 E+00 μCi/cc HRM OR 1.6 E-09 amps HRM	1.93 E+00 μCi/cc HRM OR 7.8 E-10 amps HRM	7.92 E+03 cps LRM
Turbine Bldg RAGEMS	5.0 E-01 μCi/cc HRM OR 2.0 E-10 amps HRM	2.51 E+05 cpm LRM	8.11 E+04 cpm LRM	8.11 E+02 cpm LRM

EXAMINATION ANSWER KEY

ILT 09-1 NRC SRO Exam



5-Mile Radius from Oyster Creek

Note that each radiological release listed below has been at the value indicated for > 15 minutes.

	Release	Wind Direction
A.	Main Stack RAGEMS 6.6 E+01 μCi/cc HRM	90 °
B.	Main Stack RAGEMS 4.3 E+00 μCi/cc HRM	90 °
C.	TB RAGEMS 5.9 E-01 μCi/cc HRM	270 °
D.	TB RAGEMS 2.78 E+05 cpm LRM	270 °

Answer: C

Answer Explanation:

QID: 09-1 NSRO23		
Question # / Answer	23	Developer/Date: NTP 1/10/10

Knowledge and Ability Reference Information					
K&A				Importance Rating	
				RO	SRO
Radiation Control 2.3.15 - Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.					3.1
Level	SRO	Tier	3	Group	
General References	EP-AA-1010	EP-AA-112-100-F-01		EP-AA-111-F-10	

Explanation	The plant was at rated power when an event occurred resulting in an offsite release. The question asks what the smallest release rate would require the Shift Emergency Director to recommend evacuation of Point X. Evacuation and sheltering are components of Protective Action recommendations (PARs). Thus, the question asks when a PAR will be recommended. PARs are recommended at the General Emergency level. Point X on the map is in the eastern section. For wind to blow in this section, it must come from a westerly direction. Indicated wind direction is "from", not "to". A wind from 270 degrees would require evacuation of sections ENE, E, & ESE. Point X is contained in the E section. Therefore, a GE must be declared and indicated wind is from the west or 270 degrees. For plant-based PARs, evacuation of a 2-mile radius and 5 miles downwind is required, and Point X is within 5 miles downwind. Answer C is lists a release rate which would require a GE declaration and indicated wind is from the west at 270 degrees. Answer C is correct. Answer A is a GE, but the wind direction is incorrect. Answer A is incorrect. Answer B is a SAE and the incorrect wind direction. Answer B is incorrect. Answer D is a SAE and the correct wind direction. Answer D is incorrect. If the candidate confuses that emergency level which requires PAR determination and the indicated wind direction, all answers are plausible.

References to be provided during exam:	**None**	
Learning Objective	G-101 DBIG LO G-101 DBIG 01	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:SPK**
	NUREG 1021 Appendix B: Solve a problem using knowledge and its meaning.			
10CRF55 Content	**55.41**		**55.43**	**2**
	(SRO Only) Conditions and limitations in the facility license			

Time to Complete: 1-2 minutes

24 **ID: 09-1 NSRO24** **Points: 1.00**

The plant was in a refuel outage. Due to the loss of SRM 24, fuel was being shuffled in core quadrants 1, 2, and 3.

While reviewing work packages for the following day, you note a maintenance activity requiring a tagout to de-energize 24 VDC Power Panel A.

If the maintenance activity were allowed to occur as scheduled, which of the following states the impact on refueling, if any?

A. There will be **no** impact on the core alterations.

B. Core alterations will be restricted to core quadrant 3 **only**.

C. **All** core alterations must cease due to the loss of the required number of operable SRMs.

D. **All** core alterations must cease due to the loss of Secondary Containment Integrity and the auto start of SGTS.

Answer: C

Answer Explanation:

QID: 09-1 NSRO24		
Question # / Answer	24	Developer/Date: NTP 1/11/10

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
Equipment Control 2.2.36 - Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.						4.2
Level	SRO	Tier	3	Group		
General References	TS 3.9		GE 706E812 sh. 3, 5, 6			

Explanation	The plant is in a refuel outage with SRM 24 inoperable. Core alterations are occurring in the other core quadrants with operable SRMs. If 24 VDC Power Panel A is de-energized, this will render SRMs 21 and 22 inoperable. TS 3.9.D provides the following: During CORE ALTERATIONS at least two (2) source range monitor (SRM) channels shall be OPERABLE and inserted to the normal operating level. One of the OPERABLE SRM channel detectors shall be located in the core quadrant where CORE ALTERATIONS are being performed, and another shall be located in an adjacent quadrant. TS 3.9.G provides the following: With any of the above requirements not met, cease CORE ALTERATIONS or control rod removal as appropriate, and initiate action to satisfy the above requirements. Since only 1 SRM remains operable in quadrant 3, the requirement for 2 operable SRMs will not be met and core alterations must cease. Answer C is correct. Since the refuel activities are impacted, answer A is incorrect. Since SRM 23, in core quadrant 3 is still operable, the candidate may think that fuel moves are still allowed in that single quadrant. But as shown, 2 SRMs are required. Answer B is incorrect. The loss of 24 VDC Power Panel will isolate RB normal Vent and initiate the Standby Gas treatment System (SGTS). This will not cause SGTS or Secondary Containment to be inoperable. Answer D is incorrect. 4/9/10: This question was modified following NRC feedback.

References to be provided during exam:	**Reference Deleted**	
Learning Objective	2621.828.0.0029 LO 215-10451	

Question Source (New, Modified, Bank)			**New**	
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	**X 3:PEO**
	NUREG 1021 Appendix B: Predict an event or outcome			
10CRF55 Content	**55.41**		**55.43**	**2**
	(SRO Only) Facility operating limitations in the tech Specs and the bases			

Time to Complete: 1-2 minutes

25 **ID: 09-1 NSRO25** **Points: 1.00**

The plant was at rated power when the Shift Manager declared an Unusual Event, EAL HU6, due to a fire.

15 minutes later, the Shift Emergency Director declared an Alert, EAL HA6, due to worsening conditions from the fire.

IAW EP-AA-112-100-F-01, Shift Emergency Director Checklist, which of the following is required at the **new** EAL which is **not** required at the original EAL?

A. Initiation of a site evacuation.

B. Notification of state/local authorities within 15 minutes.

C. Activation of the Emergency Response Organization (ERO).

D. Determination of Protective Action Recommendations (PARs).

Answer: C

Answer Explanation:

QID: 09-1 ASRO25		
Question # / Answer	25 C	Developer/Date: NTP 8/27/09

Knowledge and Ability Reference Information						
K&A					Importance Rating	
					RO	SRO
2.4.29 Knowledge of the emergency plan						4.4
Level	SRO	Tier	3	Group		
General References	EP-AA112-100-F-01	EP-AA-1010				

Explanation	The question shows that an Unusual Event emergency was declared and then upgraded to an Alert. IAW the Shift Emergency Director Checklist, activation of the emergency response organization (ERO) is required at the alert level or higher (except a security event). Answer C is correct. Initiation of a site evacuation is performed at the Site Area Emergency or higher. Answer A is incorrect. Notification of state/local authorities is performed at all emergency levels. Answer C is incorrect. PARs are only recommended at the General Emergency level. Answer D is incorrect.
References to be provided during exam:	**None**
Learning Objective	G-101 DBIG LO G-101 DBIG 01

Question Source (New, Modified, Bank)				**New**
Cognitive Level	**Memory or Fundamental Knowledge**	**X 1:P**	**Comprehension or Analysis**	
	NUREG 1021 Appendix B: Procedure steps and cautions			
10CRF55 Content	**55.41**		**55.43**	**1**
	(SRO Only) Conditions and limitations in the facility license			
Time to Complete: 1-2 minutes				