EXAMINATION ANSWER KEY

2016 RO NRC TEST

1 **ID: 1248364** **Points: 1.00**

The plant is at rated conditions in a normal electric plant lineup with the following:

- Time = 0 seconds: A LOCA occurs.
- Time = 60 seconds: RPV water level is 80" TAF and lowering.
- Time = 75 seconds: Breaker S1B fails to auto close
- Time = 85 seconds: EDG #2 DISABLE annunciator alarms due to low cooling water pressure.

Based on these conditions, what is the response of the EDGs at 100 seconds?

	EDG #1	EDG #2
A.	Fast started	Idle started and tripped
B.	Fast started	Idle started and loaded
C.	Idle started	Fast started to 900 RPM and tripped
D.	Idle started	Fast started to 900 RPM and loaded

Answer: D

Answer Explanation			
K&A	264000 EDGs K1.04 Knowledge of the physical connections and/or cause- effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following: Emergency generator cooling water system (3.2/3.3)		
Level: RO	**Tier:** 2		**Group:** 1
References	RAP-T4F	EDG OP 341	
Explanation	Proposed Answer: D The candidate must recognize that the #1 EDG will idle start due to the LOCA. Since S1B failed to repower Bus 1B which powers Bus 1D, the #2 EDG received a fast start signal due to undervoltage. The fast start signal bypasses the low cooling water pressure protection and continues to run loaded to the bus. A is Incorrect. Plausible – Both EDGs received a start signal on the low RPV water level. However that signal is an idle start signal, not a fast start. EDG #2 also received a fast start signal on undervoltage when S1B failed to close and re-energize the bus. B is Incorrect. Plausible – Both EDGs received a start signal on the low RPV water level. However that signal is an idle start signal, not a fast start. EDG #2 also received a fast start signal on undervoltage when S1B failed to close and re-energize the bus. C is Incorrect. Plausible – EDG #1 received only an idle start signal. EDG #2 received a fast start signal and accelerated to 900 RPM, however, the fast start logic bypasses the low cooling water pressure signal so EDG #2 will not trip.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Lesson Plan Learning Objective/	N-OC-2621.828.0.013 Emergency Diesel Generators EDG-00813 - Explain the differences between normal EDG start sequence and fast start sequence, including trip bypasses and automatic fault resets.					
References Provided	**ILT:** None					**LORT:** Open
Question Source (New, Modified, Bank)	Modified					
Previous 2 NRC Exams (ILT Only)	No					
Cognitive Level	**Memory or Fundamental Knowledge**				**Comprehension or Analysis**	X
10CFR55 Content	**1b**	7	**55.43b**			
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.					
Justification for LORT K&A <3.0	N/A					
Time to Complete:	1-2 minutes					
Point Value:	1					
System ID No.:	0	**PRA:**		No		
Safety Function(s):	6	☒ **ILT**				
Category(s) (LORT Only):	/A	☐ **LORT**				

EXAMINATION ANSWER KEY

2016 RO NRC TEST

2 **ID: 1248366** **Points: 1.00**

The plant is operating at approximately 95% power with the following conditions:

- Operators are completing ABN-40 actions for a stuck open EMRV, which is now CLOSED.
- The unit RO fails to zero the deviation on the Master Feedwater Controller prior to returning it to AUTO
- RPV water level is rising and has been greater than 182" TAF for 5 seconds on RE05/19A and RE05/19B.

Which one of the following automatic actions occurs as a **direct result** of the high RPV water level condition sensed on RE05/19A & RE05/19B

A. MSIVs close

B. All operating MFPs trip

C. Isolation Condensers go into service

D. "A", "B", & "C" Main Feed Regulating valves fully close

Answer: B

<table>
<tr><th colspan="5">Answer Explanation</th></tr>
<tr><td>K&A</td><td colspan="4">259002 Reactor Water Level Control System
K1.01 Knowledge of the physical connections and/or cause effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: RPS (3.8/3.9)</td></tr>
<tr><td>Level: RO</td><td colspan="2">Tier: 2</td><td colspan="2">Group: 1</td></tr>
<tr><td>References</td><td>RAP-H5d</td><td>619.3.013</td><td colspan="2"></td></tr>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: B

Explanation: RPS level instruments RE05/19A and B provide input to the Reactor Overfill Protection System (ROPS). The ROPS functions to trip all operating RFPs if a high reactor water level condition (>181") is sensed on BOTH RE05/19A & RE05/19B, provided that the ROPS is not bypassed by either the switch on panel 4F or a low total feedwater flow. In the conditions provided the total feedwater flow at 95% power is > 2.23 E6 lbm/hr, so therefore ROPS is NOT bypassed.

A. Plausible - A Turbine Trip will have occurred resulting in Stop Valve closures, not MSIVs. MSIV closure a protective function of RPS and may eventually occur, however it is not a direct result of the given conditions.
C. Plausible – This is a protective function of RPS and may eventually occur, however it is not a direct result of the given conditions.
D. Plausible – This would have a similar effect of preventing RPV overfill, however overfill protection is accomplished via tripping the FWPs, vice closing the FRVs. Also, FWLC is sensed off of GEMAC detectors & candidate must know that FRV's control signals are not received from RE05/19A & RE05/19B</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="4">2621.828.0.0018 - Feedwater Control System
FWC-10444 - Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td>LORT: Open</td></tr>
</table>

EXAMINATION ANSWER KEY

Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	259002	**PRA:**	No	
Safety Function(s):	2	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

3 **ID: 1248367** **Points: 1.00**

Unit Substation, USS 1B2, de-energized due to a fault.

Which one of the following describes the response of the Isolation Condensers (ICs) to automatic initiation and isolation signals?

	Response to Automatic initiation signal	Response to Automatic isolation signal
A.	Both ICs initiate	All IC isolation valves close
B.	Both ICs initiate	NOT all IC isolation valves close
C.	Only IC 'A' initiates	All IC isolation valves close
D.	Only IC 'A' initiates	NOT all IC isolation valves close

Answer: B

Answer Explanation				
K&A	207000 Isolation (Emergency) Condenser K2.01 Knowledge of electrical power supplies to the following: Motor operated valves: BWR-2,3 (3.6/3.8)			
Level: RO	**Tier:** 2		**Group:** 1	
References	307	ABN-48		
Explanation	Proposed Answer: B Explanation: USS 1B2 supplies MCC 1B21 which supplies power to V-14-32, the AC steam IV for IC 'B', which is normally open. The loss of the AC bus will not prevent V-14-35, The DC Condensate Return from 'B' Condenser from opening on an initiation signal. On an isolation signal, the normally open V-14-32 will fail to close as required. A. Plausible – Both ICs will initiate, however V-14-32 will fail to isolate due to the loss of power. If the applicant believes that V-14-32 is a normally closed valve since power is lost then all isolation valves would be closed on an isolation signal. C. Plausible if the candidate believes that since V-14-32 has no power then IC 'B' will not initiate becuase they believe V-14-32 is a normally closed valve and therefore only IC 'A' will initiate. Also if the candidate does not understand V-14-32 is a normally open valve and believes the valve is already closed therefore all IC isolation valves will close on an isolation signal. D. Plausible if the candidate believes that since V-14-32 has no power then IC 'B' will not initiate becuase they believe V-14-32 is a normally closed valve and therefore only IC 'A' will initiate. Plausible if the candidate does not understand V-14-32 is a normally open valve.			
Lesson Plan **Learning Objective/**	2621.828.0.0023 - ISOLATION CONDENSERS ICS-2030 - DESCRIBE the Isolation Condenser design feature(s) and/or interlocks (including signals and setpoints) which provide for the following: Automatic system initiation, Automatic system isolation			
References Provided	**ILT:** None		**LORT:** Open	

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Question Source (New, Modified, Bank)	NEW				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	207000	**PRA:**	No		
Safety Function(s):	4	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

4 **ID: 1248368** **Points: 1.00**

The plant is at rated power with the following air compressor lineup:

- 1-1 air compressor is the **LAG** compressor
- 1-2 air compressor is the **LEAD** compressor
- 1-3 air compressor is in Standby

The following annunciators then alarm:

- 1A1 MN BRKR TRIP
- 1A1 MN BRKR OL TRIP

Which one of the following states the impact on the Instrument Air System?

Air Compressor…

A. 1-1 auto starts in **LAG** and will maintain Instrument Air pressure 95 - 110 psig.

B. 1-3 will automatically start and maintain Instrument Air pressure 85 - 105 psig.

C. 1-1 auto starts in **LAG** and will maintain Instrument Air pressure 105 - 120 psig.

D. 1-2 continues to operate in **LEAD** and will maintain Instrument Air pressure 105 - 120 psig.

Answer: D

Answer Explanation			
K&A	300000 - Instrument Air System K2.01 - Knowledge of electrical power supplies to the following: Instrument air compressor (2.8/2.8)		
Level: RO	**Tier:** 2		**Group:** 1
References	334		
Explanation	Proposed Answer: D Explanation: Air Compressor (AC) 1-1 receives power from USS 1A1. AC 1-2 and 1-3 receive power from USS 1B1. The annunciators given indicate a loss of USS 1A1 where it is not immediately available to be restored. A. Plausible – 95-110 psig is the lag compressor control band, however USS 1A1 is the power supply to AC 1-1, therefore, it is not available. It is plausible if the applicant believes AC 1-2 losses power therefore AC 1-1 would start when air pressure reaches 95 psig B. Plausible – Two instrument air compressors are powered from 1B1, and 1 compressor is powered from 1A1. If the candidate mistakenly believes 1-3 is the air compressor that still has power, then it would maintain between 85-105 psig. C. Plausible – USS 1A1 is the power supply to AC 1-1, therefore, it is not available. 105-120 psig is the correct band for the Lead compressor setting that would be maintained. It is plausible if the applicant believes AC 1-2 losses power therefore AC 1-1 would start and then confuses the lead/lag setpoints.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Lesson Plan Learning Objective/	2621.828.0.0043 CAS-10445- DESCRIBE the Isolation Condenser design feature(s) and/or interlocks (including signals and setpoints) which provide for the following: Automatic system initiation, Automatic system isolation				
References Provided	**ILT:** None			**LORT:** Open	
Question Source (New, Modified, Bank)	Modified				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X	
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	300000	**PRA:**	No		
Safety Function(s):	8	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

5 **ID: 1248369** **Points: 1.00**

The plant is operating at rated conditions with Cleanup Recirc Pump B in service.

Then, DC Bus B de-energizes due to an electrical fault.

Which one of the following describes how the Reactor Water Cleanup System is affected?

A. Loss of Indication and Control Power, ONLY

B. Cleanup Pump B trips off, ONLY

C. All four System Isolation valves shut, ONLY

D. No impact on RWCU

Answer: A

Answer Explanation				
K&A	263000 - D.C. Electrical Distribution K3.02 - Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Components using D.C. control power (i.e. breakers) (3.5/3.8)			
Level: RO	**Tier:** 2		**Group:** 1	
References	ABN-54			
Explanation	Proposed Answer: A Explanation: IAW ABN-54, Cleanup Pump B receives a trip signal on a loss of DC-B, but will remain running due to a loss of tripping power. Indication and Control Power is lost. B. Plausible because Cleanup Pump B would trip off on a loss of DC-B. Also, Cleanup Pump B does receive a trip signal, but does not trip due to loss of tripping power. C. Plausible because Cleanup isolation valves are impacted, however it is a loss of indication, not an isolation signal. D. Plausible because the RWCU system continues to operate with no trips or isolations, however indication and control power are lost.			
Lesson Plan Learning Objective/	2621.828.0.0039 - REACTOR WATER CLEANUP SYSTEM RCU-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**	

EXAMINATION ANSWER KEY

2016 RO NRC TEST

10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	263000	**PRA:**	No	
Safety Function(s):	6	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

6 **ID: 1248370** **Points: 1.00**

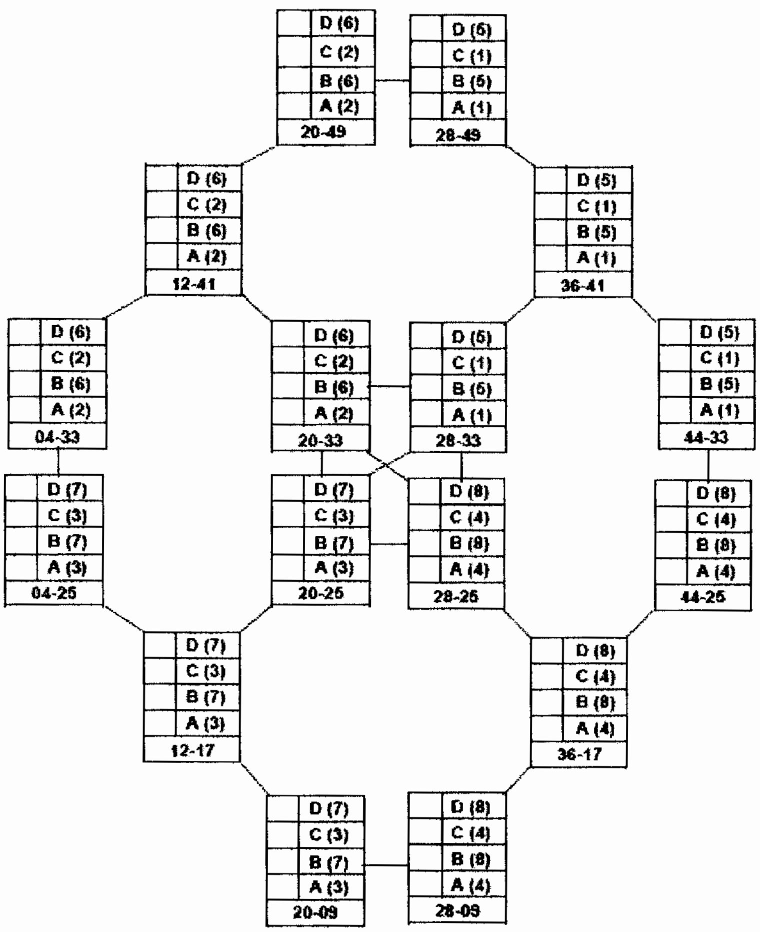
The plant is operating approximately 35% power with the following:

- A plant shutdown in progress
- The following LPRMs are bypassed: 20-09B, 28-33B, 28-49D, 44-33D
- There is a blown fuse and power was lost on RPS system 1
- Subsequently, LPRM 36-41B fails downscale
- The operator bypasses LPRM 36-41B

Which one of the following describes the plant response (if any) for (1) the failure of LPRM 36-41B and (2) if the operator bypasses LPRM 36-41B APRM input?

ATTACHMENT 403-2 LPRM AND APRM STATUS INFORMATION SHEET

NOTE
BYPASSED OR INOPERABLE LPRM'S/APRM'S SHOULD BE MARKED WITH AN "X", THOSE LPRMs WHICH MAY NOT BE BYPASSED SHOULD BE MARKED WITH AN "O" AND THOSE OPERABLE LPRMs IN AN APRM CHANNEL THAT IS BYPASSED SHOULD BE MARKED WITH AN "→"



APRMs: 1 2 3 4 5 6 7 8

SRO approved: Date Time

	(1)	(2)
A.	No response	½ Scram
B.	No response	Full Scram
C.	LPRM 36-41B amber light on 4F illuminates	½ Scram
D.	LPRM 36-41B amber light on 4F illuminates	Full Scram

Answer: D

Answer Explanation			
K&A	212000 - Reactor Protection System K3.03 - Knowledge of the effect that a loss or malfunction of the REACTOR PROTECTION SYSTEM will have on following: Local power range monitoring system: Plant-Specific (3.3/3.4)		
Level: RO	**Tier:** 2		**Group:** 1
References	403	RAP-G7f	
Explanation	Proposed Answer: D Explanation: Three LPRMs assigned to APRM 5 (RPS system 2) are bypassed (28-33B, 28-49D, and 44-33D). When LPRM 36-41B, which also inputs to APRM 5 (RPS system 2), fails downscale, an associated amber light is illuminated on Panel 4F. When the operator bypasses the failed LPRM, it results in <5 inputs to APRM 5, therefore a half scram signal is generated for an INOP condition on APRM 5. Since RPS 1 has already lost power and there is a ½ scram signal in, when the LPRM is bypassed there are <5 inputs for RPS system 2 creating RPS 2 to provide a scram signal therefore a full scram would occur. KA Match Justification: A Blown fuse represents a malfunction in the RPS System. This impacts the LPRM system in that a failure of an LPRM could drive the plant into a full scram condition. A. Plausible if the candidate is not aware of the impact of an LPRM failure. An alarm would occur initially. B. Plausible if the candidate is not aware of the impact of an LPRM failure, alarm would occur initially. The half scram that is generated would cause all rods to insert because of the failure of RPS 1. C. Plausible if the candidate doesn't know the inputs to APRM 5 which is RPS system 2 and doesn't realize there are <5 there would only be a ½ scram in.		
Lesson Plan Learning Objective/	2621.828.0.0029 - NUCLEAR INSTRUMENTATION NIS-10444 - Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components		
References Provided	**ILT:** None		**LORT:** Open
Question Source (New, Modified, Bank)	Modified from bank NI-23		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	212000	**PRA:**	No	
Safety Function(s):	7	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

7 **ID: 1248371** **Points: 1.00**

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

- A fault occurs on 4160V Bus 1B
- EDG 2 re-energizes its respective BUSSES

Which one of the following describes the response of Automatic Transfer Switch (ATS) IT-3 to this electrical transient?

A. loads are automatically transferred to VMCC-1A2. ATS IT-3 loads remain on this source until manually re-transferred.

B. loads are automatically transferred to the VMCC-1A2. ATS IT-3 loads automatically transfer back after power restoration.

C. inverter is automatically supplied power from DC Distribution Center B. CIP-3 Rotary inverter remains supplied from this source until manually re-transferred.

D. inverter is automatically supplied power from DC Distribution Center B. CIP-3 Rotary inverter power supply automatically transfers back after power restoration.

Answer: D

Answer Explanation			
K&A	262002 - Uninterruptable Power Supply (A.C. /D.C.) K4.01 - Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following: Transfer from preferred power to alternate power supplies (3.1/3.4)		
Level: RO	**Tier:** 2		**Group:** 1
References	ABN-48	339	RAP-9XF5c
Explanation	Proposed Answer: D Explanation: Loss of 4160V Bus 1B causes a loss of power to VMCC1B2. This causes a loss of the normal AC power supply to the rotary inverter for ATS IT-3. The rotary inverter transfers to the DC power supply. On subsequent return of AC power, when the EDG re-energizes its respective busses, the rotary inverter transfers back to AC drive. A. Plausible – VMCC-1A2 is an alternate power supply to IT-3, however the conditions do not exist in the stem that would result in power transfer to VMCC-1A2 B. Plausible – VMCC-1A2 is an alternate power supply to IT-3, however the conditions do not exist in the stem that would result in power transfer to VMCC-1A2 C. Plausible – The inverter does transfer to the DC Bus, however, it will transfer back upon return of AC power source because the control switch is in AUTO-Run. If the control switch was in DC-Run then the rotary inverter would stay on the DC drive. Operating the control switch in DC-run mode is only ran during testing and not during normal operation of the rotary inverter.		
Lesson Plan **Learning Objective/**	2621.828.0.0056- VITAL AC DISTRIBUTION SYSTEM VAC-10438 - Using the system P&IDs, locate each of the system components and explain its operation and limitations within the system. VAC-10441 - Given the system logic/electrical drawings, describe the system trip signals, setpoints and expected system response including power loss or failed components.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

References Provided	**ILT:** None			**LORT:** Open
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**	
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	262002	**PRA:**	No	
Safety Function(s):	6	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

8 **ID: 1248372** **Points: 1.00**

Which one of the following is used to offset LPRM detector aging?

The LPRM detector...

A. flux amplifier gain can be adjusted during routine calibrations.

B. high voltage power supply can be lowered to off-set the buildup of Plutonium.

C. ion chamber is coated with enriched U-235 for a service life of at least six (6) years.

D. strings are periodically rotated between high and low flux areas during refueling outages.

Answer: A

Answer Explanation				
K&A	215005 - Average Power Range Monitor/Local Power Range Monitor K4.06 - Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Effects of detector aging on LPRM/APRM readings (2.6/2.8)			
Level: RO	**Tier:** 2		**Group:** 1	
References	403	620.3.009		
Explanation	Proposed Answer: A Explanation: Increasing the gain in the flux amplifier is the way to offset detector aging which is brought about by burnup of uranium and gas equalization in the detector. B. Plausible – Decreasing the value of high voltage is an allowable adjustment that can be made, however it would decrease instrument sensitivity and not offset detector aging. C. Plausible – U-235 is there to allow the detector to work, not to offset aging. U-234 is added to allow for conversion to U-235 to offset aging. D. Plausible – LPRM strings are routinely replaced during refueling outages. They are not rotated.			
Lesson Plan Learning Objective/	2621.828.0.0029 - NUCLEAR INSTRUMENTATION NIS-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**	
10CFR55 Content	**55.41b**	7	**55.43b**	

10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	215005	**PRA:**	No	
Safety Function(s):	7	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

EXAMINATION ANSWER KEY

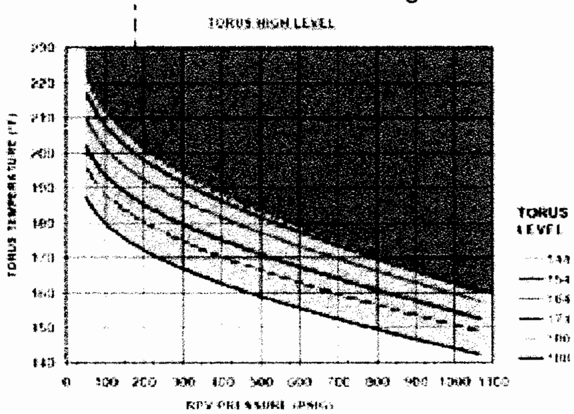
2016 RO NRC TEST

9 **ID: 1248373** **Points: 1.00**

The plant is experiencing a Torus leak with the following:

- The reactor is manually scrammed
- Torus water level is 90" and lowering
- Torus Water Temperature is 110F and rising slowly
- Torus makeup is in progress using Core Spray.
- An Emergency Depressurization is in progress
- Reactor Pressure is 500 psig and lowering
- DW Pressure is 2.0 psig and steady
- Torus pressure is 2.0 psig and steady

Which one of the following will occur **next**?



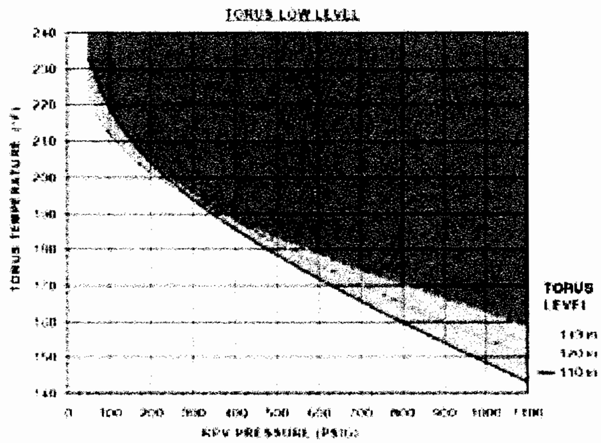


FIG. F HCTL
HEAT CAPACITY TEMPERATURE LIMIT

A. Torus to drywell vacuum breakers will open.

B. Torus water temperature will exceed HCTL.

C. The Torus air space will rapidly pressurize.

D. Drywell Downcomer openings will begin to uncovered.

Answer: C

Answer Explanation				
K&A	239002 - Safety Relief Valves K5.05 - Knowledge of the operational implications of the following concepts as they apply to RELIEF/SAFETY VALVES: Discharge line quencher operation (2.6/2.9)			
Level: RO	**Tier:** 2		**Group:** 1	
References	EOP Users Guide	EMG-SP12		

Explanation	Proposed Answer: C Explanation: The EOP Users Guide states, "The EMRVs may only be opened when Torus level is above their discharge device (90 in.). This ensures steam suppression and will prevent directly pressurizing the Torus air space which could lead to Primary Containment failure." With Torus water level at 90" and lowering, this is the next immediate concern. A. Plausible – The Torus to DW vacuum bkrs will open on rising D/P however the Torus pressure must be higher than the drywell to Open. If the applicant believes the vacuum bkrs open on a higher torus pressure this is plausible distractor but it will not happen next as the EMRV lines will become uncovered next then pressure will rise after that to open the vacuum bkrs. B. Plausible – Torus temperature is elevated and rising. HCTL is a concern, however, not as immediate as uncovering the EMRV discharge piping. D. Plausible – Uncovering the downcomers is a concern, however they are already uncovered based on the conditions given. EOP users guide states an emergency depressurization is required prior to 110" in the torus for that reason.Technical Reference(s): EOP Users Guide, EMG-SP37, EMG-SP12			
Lesson Plan Learning Objective/	2621.845.0.0056 - PRIMARY CONTAINMENT CONTROL PCC-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	8	**55.43b**	
10CFR55 Explanation	Component, capacity, and functions of emergency systems			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	239002	**PRA:**	No	
Safety Function(s):	3	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

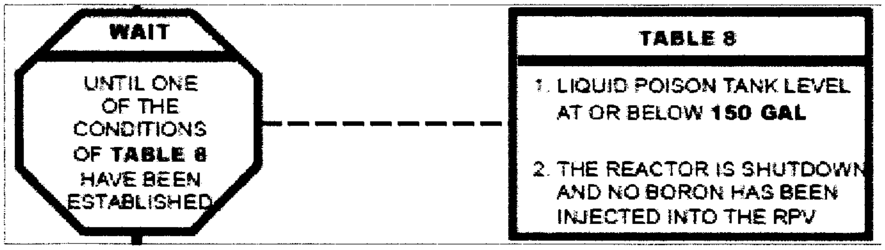
EXAMINATION ANSWER KEY

2016 RO NRC TEST

10 **ID: 1248374** **Points: 1.00**

A plant transient has occurred resulting in entry to the RPV Control - With ATWS EOP.

Before a cooldown can begin during the ATWS, an EOP step states:



Which one of the following is the basis for liquid poison tank level at or below 150 gallons **AND** the operational implication correlating to this boron weight for this step **ONLY**?

	Basis of 150 gallons	Operational implication
A.	Cold Shutdown Boron Weight (CSBW) injected	Cooldown may be performed even if control rod insertion is NOT sufficient to shut down the reactor.
B.	Hot Shutdown Boron Weight (HSBW) injected	Cooldown may be performed even if control rod insertion is NOT sufficient to shut down the reactor.
C.	Cold Shutdown Boron Weight (CSBW) injected	Cooldown may NOT be performed unless all control rods are inserted to or beyond position 04.
D.	Hot Shutdown Boron Weight (HSBW) injected	Cooldown may NOT be performed unless all control rods are inserted to or beyond position 04.

Answer: A

Answer Explanation				
K&A	211000 – Standby Liquid Control System K5.03- Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM: Shutdown Margin (3.2/3.5)			
Level: RO	**Tier:** 2		**Group:** 1	
References	EOP Users Guide			

<table>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: A

Explanation: The basis of this step is to either ensure Cold shutdown boron weight has been injected or all rods are in. Under ATWS conditions, a cooldown to cold shutdown conditions may be initiated only if (1) the reactor is shutdown and no boron has been injected, or (2) Cold Shutdown Boron has been injected. If no boron has been injected into the RPV, the cooldown may be performed if control rod insertion is sufficient to shut down the reactor, even if the shutdown margin is small. A return to criticality under these conditions is acceptable since terminating the cooldown will stop the power increase.
If any amount of boron less than the cold shutdown amount has been injected, cooldown is not permitted unless it can be determined that the reactor will remain shutdown under all conditions without the boron.
An RPV depressurization and cool down adds positive reactivity to the core due to decreasing moderator temperature. Under conditions where several control rods have not been fully inserted, this cool down could result in recriticality. If CSBW has been injected into the RPV, recriticality will not occur due to cooldown. Per EOP Users Guide, " Initiation of a depressurization with any amount of boron less than the Cold Shutdown Boron Weight is not permitted unless it has been determined that the Reactor will remain shutdown on control rods alone."

B. Plausible if the candidate does not know the value in the step represents CSBW vice HSBW and that if HSBW is injected the cannot cooldown unless the rod insertion is sufficient.
C. Plausible in that this step does apply to CSBW but a cooldown can be performed once the CSBW is injected into the RPV by this step. If the applicant believes that since the rods are not at position 04 then they are still in an ATWS condition and does not recall the basis of the CSBW then it is plausible that a cooldown should not be performed. It is false however because the CSBW has been injected into the core therefore a cooldown can commence regardless of rod position.
D. Plausible in that it is true that the reactor cannot be cooled down since all rods are not at position 04 or beyond and only the HSBW has been injected. The basis for this step applies to CSBW as indicated by the 150 gallons in the liquid poison tank. If the applicant does not recall that the 150 gallons applies to CSBW injection vise HSBW then this distractor is plausible.</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="4">2621.845.0.01B - RPV CONTROL-WITH ATWS
EWA-03055 - Given a copy of RPV Control, describe in detail each step or conditional statement, including technical basis, and how to perform each step as required.</td></tr>
<tr><td>References Provided</td><td>ILT: None</td><td colspan="2"></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">New</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td>X</td><td>Comprehension or Analysis</td><td></td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>6</td><td>55.43b</td><td></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Design, components, and functions of reactivity control mechanisms an instrumentation.</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	211000	**PRA:**	No	
Safety Function(s):	1	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

11 **ID: 1248375** **Points: 1.00**

A plant startup is in progress with the following:

- The Mode Switch is in STARTUP with control rod withdrawal in progress
- IRMs 11, 12, 15, 16, and 18 read approximately 75% out of 125% on Range 2
- IRMs 13, 14, and 17 read approximately 15% out of 125% on Range 3

Then, a malfunction in the IRM drive circuitry causes the IRM 13 detector to withdraw to the full-out position.

Which one of the following states the effect on the plant AND the required operator actions to continue withdrawing control rods?

This will result in panel annunciators...

A. ONLY; withdrawing control rods may continue without any other control panel manipulations.

B. and a rod block from IRM downscale ONLY; bypassing IRM 13 is required to continue withdrawing control rods.

C. and a rod block from IRM downscale AND IRM detector position; bypassing IRM 13 is required to continue withdrawing control rods.

D. and a rod block from IRM detector out of position and cannot be bypassed and the startup can not continue.

Answer: C

<table>
<tr><th colspan="4">Answer Explanation</th></tr>
<tr><td>K&A</td><td colspan="3">215003 – Intermediate Range Monitor System
K6.04- Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : Detectors (3.0/3.0)</td></tr>
<tr><td>Level: RO</td><td colspan="2">Tier: 2</td><td>Group: 1</td></tr>
<tr><td>References</td><td>402.2</td><td>RAP-H7a,</td><td></td></tr>
<tr><td>Explanation</td><td colspan="3">Proposed Answer: C

Explanation: The detector moving out of the core will cause IRM 13 to go downscale as the detector moves out of the core. Additionally, a rod block will be caused by the detector not being fully inserted with the Mode Switch in STARTUP. IRM 13 must be bypassed to permit clearing the rod block and continuing the startup.

A. Plausible – It will result in panel annunciators, but also a rod block. Plausible if the candidate does not know IRM rod block setpoints.
B. Plausible – IRM downscale will generate a rod block. In addition to IRM Downscale a rod block will be caused by the detector not being fully inserted with the Mode Switch in STARTUP.
D. Plausible – There is a rod block, however the detector can be bypassed</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="3">2621.828.0.0029 NUCLEAR INSTRUMENTATION
LO NIS-10444 Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

References Provided	**ILT:** None			**LORT:** Open	
Question Source (New, Modified, Bank)	New				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X	
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes an automatic and manual features				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	215003	**PRA:**	No		
Safety Function(s):	7	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

12 **ID: 1248376** **Points: 1.00**

The plant was at rated power with the STANDBY GAS SELECT switch in SYS 2, when the following radiation monitoring annunciator alarmed:

- AREA/VENT DNSCL

Investigation revealed that REACTOR BUILDING VENT MANIFOLD NO. 1 radiation monitor indicates downscale.

Which of the following states the impact on the Standby Gas treatment System (SGTS)?

A. **BOTH** SGTS Fans are in standby and **BOTH** can auto start.

B. **BOTH** SGTS Fans have auto started and will remain running.

C. **ONLY** SGTS Fan 2 has auto started and will remain running.

D. **BOTH** SGTS Fans have auto started and SYS 1 fan will shutdown after a time delay.

Answer: A

Answer Explanation			
K&A	261000 – Standby Gas Treatment System 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
References	RAP-10F4g	420	

<table>
<tr><td rowspan="1">Explanation</td><td colspan="4">Proposed Answer: A

Explanation: The question stem describes a downscale indication of the #1 RB vent manifold radiation monitor (of which there are 2). The logic for SGTS auto initiation is for either vent manifold radiation monitor to exceed the upscale trip point. When this occurs, both SGTS fans start. When it has been assured that the selected fan is functioning properly, the secondary fan will auto secure after a time delay. The impact of a single vent manifold radiation monitor downscale failure is, there is none. The SGTS remains in standby and will auto initiate as designed when the operable radiation monitor detects an upscale trip.

The logic for SGTS auto start is independent of which radiation monitor senses an upscale trip to start both SGTS fans – radiation monitor #1 (2) is not dedicated to the auto start of SGTS fan #1 (#2).

B. Plausible – Candidate must know a downscale failure will not cause an initiation signal and The logic for SGTS auto initiation is for either vent manifold radiation monitor to exceed the upscale trip point. When this occurs, both SGTS fans start. When it has been assured that the selected fan is functioning properly, the secondary fan will auto secure after a time delay making this answer incorrect.
C. Plausible – Candidate must know a downscale failure will not cause an initiation signal and The logic for SGTS auto initiation is for either vent manifold radiation monitor to exceed the upscale trip point. When this occurs, both SGTS fans start. When it has been assured that the selected fan is functioning properly, the secondary fan will auto secure after a time delay.
D. Plausible – Candidate must know a downscale failure will not cause an initiation signal and The logic for SGTS auto initiation is for either vent manifold radiation monitor to exceed the upscale trip point. When this occurs, both SGTS fans start. When it has been assured that the selected fan is functioning properly, the secondary fan will auto secure after a time delay.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.828.0.0042 – Secondary Containment and SGTS
SGT-10441- Given the system logic/electrical drawings, describe the system trip signals, setpoints and expected system response including power loss or failed components.</td></tr>
<tr><td>References Provided</td><td>ILT: None</td><td></td><td colspan="2">LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">bank</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td>X</td><td>Comprehension or Analysis</td><td></td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>7</td><td>55.43b</td><td></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes an automatic and manual features</td></tr>
<tr><td>Justification for LORT K&A <3.0</td><td colspan="4">N/A</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	261000	**PRA:**	No	
Safety Function(s):	9	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

13 **ID: 1248377** **Points: 1.00**

The plant was operating at 100% power when the following occured:

- ADS automatically initiated and only 'A' EMRV opened
- Reactor pressure is currently 900 psig and slowly lowering.

Which one of the following ranges contains the expected maximum indication for 'A' EMRV tailpipe temperature while the valve is open?

A. <195°F

B. 195-280°F

C. 281-394°F

D. >394°F

Answer: C

Answer Explanation

K&A	218000 - Automatic Depressurization System A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the AUTOMATICDEPRESSURIZATION SYSTEM controls including: ADS valve tail pipe temperatures (3.4/3.6)		
Level: RO	**Tier:** 2		**Group:** 1
References	Steam Tables	Mollier Diagram	
Explanation	Proposed Answer: C Explanation: While Reactor coolant temperature is approximately 550°F under saturated conditions at 1025 psig, the ERV tailpipe temperature will not indicate this high. The steam passing through the ERV undergoes an isenthalpic expansion process which results in a drop in temperature at the ERV tailpipe thermocouple. The maximum expected discharge temperature is 390°F. Expansion all the way to atmospheric pressure would even result in approximately 300°F. A. Plausible if the candidate does not understand the isenthalpic expansion process. B. Plausible if the candidate does not understand the isenthalpic expansion process. D. Plausible if the candidate does not understand the isenthalpic expansion process.		
Lesson Plan Learning Objective/	2621.828.0.0026 - MAIN STEAM SYSTEM MSS-10446 - Identify and explain system operating controls / indications under all plant operating conditions.		
References Provided	**ILT:** Steam Tables /Mollier Diagram		**LORT:** Open
Question Source (New, Modified, Bank)	New		
Previous 2 NRC Exams (ILT Only)	No		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	5	**55.43b**	
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	218000	**PRA:**	No	
Safety Function(s):	3	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

14 **ID: 1248378** **Points: 1.00**

A small steam line break in the drywell has resulted in the following:

- A manual reactor scram was initiated
- RPV water level dropped to a low of 110 inches before recovering to the normal band
- Drywell pressure is 4.2 psig and slowly rising

Which one of the following describes valve(s) that would have a Green Closed Light lit, in response to an automatic isolation?

A. Isolation Condenser Vents (V-14-1, 5, 19 & 20)

B. DW Air supply valve (V-6-395)

C. Reactor recirc loop sample line valves (V-24-29 & -30)

D. N2 Makeup valves (V-23-17, 18, 19 & 20)

Answer: D

Answer Explanation			
K&A	223002 - Primary Containment Isolation System/Nuclear Steam Supply Shut-Off A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF controls including: System indicating lights and alarms (3.5/3.5)		
Level: RO	**Tier:** 2		**Group:** 1
References	EMG-SP1	RAP-C1a	
Explanation	Proposed Answer: D Explanation: Only a Containment isolation has occurred, caused by Drywell pressure greater than 3.0 psig. A Reactor isolation has not occurred because the only parameter given, RPV water level, is above the Lo-Lo isolation setpoint of 86". Therefore the N2 makeup valves will close. A. Plausible – These valves isolate on a vessel isolation of Rx water level Lo-Lo and not Rx water level Lo setpoint of 139.5 in. B. Plausible – These valves isolate on a vessel isolation of Rx water level Lo-Lo and not Rx water level Lo of 139.5 in. C. Plausible – These valves isolate on a vessel isolation of Rx water level Lo-Lo and not Rx water level Lo of 139.5 in.		
Lesson Plan Learning Objective/	2621.828.0.0032 - PRIMARY CONTAINMENT PCS-00394 - Given auto isolation signals, list or identify causes(s), system response, and affected Primary Containment System components.		
References Provided	**ILT:** None		**LORT:** Open
Question Source (New, Modified, Bank)	New		
Previous 2 NRC Exams (ILT Only)	No		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Cognitive Level	Memory or Fundamental Knowledge	X	Comprehension or Analysis		
10CFR55 Content	55.41b	5	55.43b		
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	223002	PRA:	No		
Safety Function(s):	5	☒ ILT			
Category(s) (LORT Only):	N/A	☐ LORT			

15 **ID: 1248379** **Points: 1.00**

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

The Transmission System Operator provides notification of a Voltage Reduction Alert and a potential loss of offsite power.

Based on this report, the operating crew should execute ___(1)_____.

After offsite power is lost, the operating crew should ___(2)_____.

Which one of the following completes the sentences describing the actions required for this transient assuming all automatic plant features occurred as designed?

	(1)	(2)
A.	ABN-60, Grid Emergency	Start both EDGs in accordance with Procedure 341, Emergency Diesel Generator Operation.
B.	ABN-60, Grid Emergency	Confirm all MSIV's are closed
C.	ABN-36, Loss of Offsite Power	Start both EDGs in accordance with Procedure 341, Emergency Diesel Generator Operation.
D.	ABN-36, Loss of Offsite Power	Confirm all MSIV's are closed

Answer: B

Answer Explanation			
K&A	262001 A.C. Electrical Distribution A2.11 Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Degraded system voltages (3.2/3.6)		
Level: RO	**Tier:** 2		**Group:** 1
References	ABN-60	ABN-36	

Explanation	Proposed Answer: B Explanation: With unstable grid conditions, entry into ABN-60 is required due to the Voltage reduction alert. After offsite power is lost, ABN-36 directs confirming the MSIVs are closed. A. Plausible – ABN-60 must be entered. However, the EDGs should automatically start. ABN-36 gives direction to start both EDGs in accordance with Procedure 341, Emergency Diesel Generator Operation IF they are not running, but they are expected to be running. C. Plausible if the candidate believes an potential loss of offsite power requires entry into ABN-36. ABN-36 gives direction to start both EDGs in accordance with Procedure 341, Emergency Diesel Generator Operation IF they are not running, but they are expected to be running. D. Plausible if the candidate believes an potential loss of offsite power requires entry into ABN-36				
Lesson Plan **Learning Objective/**	2621.828.0.0016 - ELECTRICAL DISTRIBUTION ACD-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, EOP and EOP support procedures, and EP Procedures.				
References Provided	**ILT:** None			**LORT:** Open	
Question Source (New, Modified, Bank)	New				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	5	**55.43b**		
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of lead changes, and operating limitations and reasons for these operating characteristics				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	262001	**PRA:**	No		
Safety Function(s):	6	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

16 **ID: 1248380** **Points: 1.00**

The plant is in COLD SHUTDOWN in preparation for a refuel outage. The following conditions exist:

- A and B Shutdown Cooling (SDC) loops are in service maintaining SDC system flow at 5000 gpm.
- C SDC loop is tagged out of service
- RPV level is 160 in TAF and steady
- B, C, & E Reactor Recirc loops are isolated; A and D Reactor Recirc loops are in service
- Rx Pressure is currently at 0 psig

A fire then occurs that trips both A and D Reactor Recirc pumps and disables the controls of all the Reactor Recirc loop valves in the original positions.

The fire has been put out but the controls for Reactor Recirc loop valves are still disabled

Which one of the following describes the resulting plant condition **AND** the required action?

	Plant Condition	Required Action
A.	SDC Flow short-cycling the core	Raise RPV water level to a minimum of >185" TAF AND Raise SDC flow to 6000 gpm
B.	SDC Flow short-cycling the core	Raise RPV water level to 170" TAF AND Raise SDC flow to 5500 gpm
C.	Thermal Stratification	Raise RPV water level to a minimum of >185" TAF AND Raise SDC flow to 6000 gpm
D.	Thermal Stratification	Raise RPV water level to 170" TAF AND Raise SDC flow to 5500 gpm

Answer: C

Answer Explanation			
K&A	205000 - Shutdown Cooling System (RHR Shutdown Cooling Mode) A2.11 - Ability to (a) predict the impacts of the following on the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation pump trips: Plant-Specific (2.5/2.7)		
Level: RO	**Tier:** 2		**Group:** 1
General References	ABN-3		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Explanation	Proposed Answer: C Explanation: IAW ABN-3, Loss of Shutdown Cooling, a certain set of conditions must be met in order to prevent temperature (thermal) stratification when the SDC system is in operation. The stem has 2 Recirc Pump operating, RPV level $\geq$ 160", and SDC in operation for decay heat removal. A trip of the operating Recirc pumps puts the plant in a condition where RPV level must be raised to >185" TAF with SDC flow rate as indicated on attachment ABN-3-2 Shutdown Cooling Operating Conditions (between 6000-6300 gpm). The stem states SDC flow is 5000 gpm therefore flow has to be raised to $\geq$ 6000 GPM to meet the flow conditions. Also since a fire has disabled recirc valve controls and the operating pumps have tripped with the loops remaining fully open then level has to be raised to a minimum of $\geq$ 185" to prevent stratification. A. Plausible if the candidate believes that short cycling will occur with no reactor recirc pumps running OR is confused about the definition of short cycling. B. Plausible if the candidate believes that short cycling will occur with no reactor recirc pumps running OR is confused about the definition of short cycling. Raising SDC flow will allow for a lower RPV water level however a SDC flow of 6000 gpm still requires and RPV level of 185" per procedure. With RPV level still at 170", this configuration would still allow for stratification to occur. D. Plausible – Raising SDC flow will allow for a lower RPV water level however a SDC flow of 6000 gpm still requires an RPV level of $\geq$185" per procedure. With RPV level still at 170" this configuration would still allow for stratification to occur.			
Lesson Plan Learning Objective/	2621.828.0.0045 - SHUTDOWN COOLING SYSTEM SDC-10447 - Given normal operating procedures and documents for the system, describe or interpret the procedural steps.			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**	
10CFR55 Content	**55.41b**	5	**55.43b**	
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of lead changes, and operating limitations and reasons for these operating characteristics			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	205000	**PRA:**	No	
Safety Function(s):	4	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

17 **ID: 1248381** **Points: 1.00**

The plant was at rated power when an event occurred. Present plant conditions are as follows:

- Drywell pressure is 3.6 psig and rising
- RPV water level is 120" and rising
- FEED PUMPS DISCHARGE PRESSURE indicates 800 psig

The Operator notes the following Core Spray System Start indications: (with no operator action)

- MAIN PUMP AMPS NZ01A indicates 50 AC AMPERES
- MAIN PUMP AMPS NZ01D indicates 0 AC AMPERES
- SYS 1 FLOW indicates approximately 100 GPM
- SYS 2 PUMP DISCH PRESS BOOSTERS indicates approximately 330 psig

Which of the following is correct regarding the observed Core Spray indications?

A. Core Spray Pump NZ01D has tripped.

B. Core Spray Pump NZ01A is running on minimum flow.

C. Core Spray System 2 is **NOT** indicating the expected discharge head.

D. Core Spray System 1 **CANNOT** provide core cooling when the RPV depressurizes.

Answer: B

Answer Explanation			
K&A	209001 - Low Pressure Core Spray System A3.04 - Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: System flow (3.7/3.6)		
Level: RO	**Tier:** 2		**Group:** 1
References	341	RAP-B1e/B2e	UFSAR 6.3.1.3.3
Explanation	Proposed Answer: B Explanation: The question stem describes the plant at power when an event resulted in a low RPV water condition and a high drywell pressure condition. Under the given conditions, core spray 1 (main pump A and booster pump a) and core spray 2 (main pump B and booster pump B) will start. With feedwater discharge pressure at 800 psig, then RPV pressure is close to this value. With core spray running at an RPV pressure > 305 psig, the core spray parallel isolation valves are closed and core spray is running on minimum flow back to the torus. This flow is approximately 100 gpm. Therefore, core spray A has started and is running on minimum flow. A. Plausible – As stated, core spray A and B start on their signals. Core spray C and D will still be in standby (off), unless a preferred core spray system fails. Since there is no indication of this in the question stem, then core spray D will be off and no amps is the expected condition – not tripped. C. Plausible – With core spray system B running on minimum flow, the discharge pressure is approximately as listed in answer C. D. Plausible – since the provided indications are the expected indications, and core spray A will provide core cooling, as designed, when RPV pressure drops < 305 psig.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Lesson Plan Learning Objective/	2621.828.0.0010 - CORE SPRAY SYSTEM CSS-10444 - Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.				
References Provided	**ILT:** None			**LORT:** Open	
Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X	
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	209001	**PRA:**	No		
Safety Function(s):	2	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

18 **ID: 1248382** **Points: 1.00**

A reactor Startup is in progress with the following:

- All SRMs are NOT fully inserted and indicate between 10^3 and 10^5 cps.
- All IRMs are in either Ranges 5 or 6.
- Then, the following malfunctions occur:
  - SRM 21 fails downscale
  - SRM 24 fails upscale

Which one of the following describes the impact of these malfunctions, if any, on the Reactor Manual Control System (RMCS)

	Impact of SRM 21 Failing Downscale	Impact of SRM 24 Failing Upscale
A.	None	Rod Block
B.	None	None
C.	Rod Block	Rod Block
D.	Rod Block	None

Answer: C

Answer Explanation			
K&A	215004 - Source Range Monitor System A3.04 - Ability to monitor automatic operations of the SOURCE RANGE MONITOR (SRM) SYSTEM including: Control rod block status (3.6/3.6)		
Level: RO	**Tier:** 2	**Group:** 1	
References	Rap-H7a		
Explanation	Proposed Answer: C Explanation: Since IRM's are not ≥ range 8 then all SRM rod blocks are still active. Since SRM 21 is not fully in and it failed downscale (less than 500 cps) then a SRM detector position rod block is received. Since SRM 24 failed upscale with IRM < range 8 then a rod block is received on SRM High greater than $1x10^5$ cps. A. Plausible – If the applicant believes that the SRM downscale rod block is bypassed with where the IRMS are at then even though SRM-21 failed downscale nothing would occur. But since the IRMs are less than range 8 the SRM downscale rod block is active. B. Plausible – If the applicant believes that the SRM rod blocks are bypassed because IRMS are in mid range then even though SRM-21 failed downscale and SRM-24 failed upscale then nothing would occur. But since the IRMs are less than range 8 the SRM rod blocks are active therefore both of them would create a rod block. D. Plausible – If the applicant believes that the SRM upscale rod block is bypassed with where the IRMS are at then even though SRM-21 failed upscale nothing would occur. But since the IRMs are less than range 8 the SRM upscale rod block is active as well as the downscale.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Lesson Plan Learning Objective/	2621.828.0.0029 - NUCLEAR INSTRUMENTATION NIS-104444 - Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	215004	**PRA:**	No	
Safety Function(s):	7	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

19 **ID: 1248383** **Points: 1.00**

The plant is operating approximately 100% power.

An unidentified leak from the RBCCW system has resulted in a loss of level in the RBCCW System Surge Tank in excess of makeup capability. The following conditions exist:

- Surge tank level is 1" in the sight glass and lowering
- RBCCW Pump 1-2 is operating
- RBCCW Pump 1-1 failed to start
- RBCCW pressure is 38 psig and lowering
- RBCCW supply temperature is 93°F and slowly rising
- Operators have been dispatched 2 minutes ago to search for the location of the leak, but it has not yet been discovered.

In accordance with ABN-19, RBCCW Failure Response, which one of the following actions are to be performed 1st?

A. Trip RBCCW Pump 1-2 and SCRAM the Reactor per ABN-1.

B. SCRAM the Reactor per ABN-1 and trip all Reactor Recirculation Pumps.

C. Trip RWCU pumps and initiate an Rapid Power Reduction.

D. Initiate a Rapid Power Reduction and trip two Reactor Recirculation Pumps.

Answer: B

Answer Explanation			
K&A	400000 - Component Cooling Water System A4.01- Ability to manually operate and / or monitor in the control room: CCW indications and control (3.1/3.0)		
Level: RO	**Tier:** 2		**Group:** 1
References	ABN-19		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

<table>
<tr><td rowspan="1">Explanation</td><td colspan="5">Proposed Answer: B

Explanation: ABN-19 states a Major unisolable RBCCW leak is defined as exceeding the makeup capacity, cannot be isolated quickly, and will result in the imminent loss of RBCCW due to loss of NPSH to the pumps. If RPV temperature is greater than 212F and a major unisolable RBCCW leak occurs, then Scram and stop all operating recirc pumps.

A. Plausible if the applicant believes tripping the remaining pump will slow the leak. However tripping the remaining pump would require a scram. There is no direction to trip the remaining RBCCW Pump. This would compound the problem.
C. Plausible – A shutdown of the RWCU system is directed later in the procedure if RBCCW temps were rising and there was not evidence in the stem that a major leak is occuring then tripping RWCU could be directed by the ABN and reducing recirc would also be directed however because operators have been dispatched over 1 minute to identify leak and there is evidence in the stem that a major unisolable leak is occuring then a scram is requied by procedure..
D. Plausible – Both of these actions will reduce the heat load on RBCCW. However, they are not directed in accordance with ABN-19 for these conditions.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="5">2621.822.0.0035 - RBCCW
RBC-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, EOP and EOP support procedures, and EP Procedures.</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td colspan="2">LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="5">New</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="5">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td colspan="2">X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>7</td><td>55.43b</td><td colspan="2"></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="5">Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features</td></tr>
<tr><td>Justification for LORT K&A <3.0</td><td colspan="5">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="5">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="5">1</td></tr>
<tr><td>System ID No.:</td><td>400000</td><td>PRA:</td><td>No</td><td colspan="2"></td></tr>
<tr><td>Safety Function(s):</td><td>8</td><td>☒ ILT</td><td></td><td colspan="2"></td></tr>
<tr><td>Category(s) (LORT Only):</td><td>N/A</td><td>☐ LORT</td><td></td><td colspan="2"></td></tr>
</table>

20 **ID: 1248384** **Points: 1.00**

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

- 1B2 MN BRKR OL TRIP

If DC-B voltage was 133 volts just prior to the event, and is lowering at a constant 2 volts/minute, then which of the following is correct? (**SEE ATTACHED**)

At the moment 1B2 tripped __(X)__ is currently **OPERABLE** and will be **INOPERABLE** in __(Y)__ minutes

BATTERY	LOAD	REQUIRED BATTERY VOLTAGE
B	* 'A' IC V-14-31	Battery Charger available
	* 'A' IC V-14-34	Battery Charger available
	* C/U Iso. Valve V-16-2	Battery Charger available
	* C/U Iso. Valve V-16-14	Battery Charger available
	CORE SPRAY NZ01C	113.3
	SERVICE WATER PUMP 1-1	111
	CRD FEED PUMP NC08B	111
	FOXBORO ER-622-120	109
	SRM/IRM	105
	EMERGENCY LIGHTING PANEL	105
	CORE SPRAY CH B (ER18B)	104
	CORE SPRAY CH A (ER18A)	103
	CONTAIN SP. RELAYS (ER8B)	102
	H2 & STATOR WATER COOLING RY.	102
	RSP RELAYS	101
	CIP-3	101
	INVERTER INV-735-001	101

	(X)	(Y)
A.	A IC V-14-34	7
B.	Core Spray NZ01C	9
C.	CRD Feed Pump NC08B	12
D.	RSP Relays	14

Answer: C

Answer Explanation			
K&A	263000 - D.C. Electrical Distribution A4.03 - Ability to manually operate and/or monitor in the control room: Battery discharge rate: Plant-Specific (2.7/2.8)		
Level: RO	**Tier:** 2		**Group:** 1
References	ABN-48		
Explanation	Proposed Answer: C Explanation: The alarm in the question stem shows a loss of USS 1B2. This results in the loss of all battery chargers to DC-A and DC-B. In 12 minutes, DC-B voltage will lower to 109 volts (133- [2x12] = 109), which is less than the minimum voltage for operability of 111 for the CRD pump. A. Plausible – The table provided shows that A IC V-14-34 is inoperable when the charger is inoperable. Thus, the valve is inoperable at the time of the initial breaker annunciator. B. Plausible – In 9 minutes, DC-B voltage will lower to 115 volts (133-[2x9] = 115), which is greater than the minimum of 113.3 volts for the pump. Thus the pump is still operable. D. Plausible – In 14 minutes, DC-B voltage will lower to 105 volts (133-[2x14]) = 105), which is greater than the minimum of 101 volts for the relays. Thus the relays are still operable.		
Lesson Plan **Learning Objective/**	2621.828.0.0012 - DC DISTRIBUTION DCD-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, EOP and EOP support procedures, and EP Procedures.		
References Provided	**ILT:** None		**LORT:** Open
Question Source (New, Modified, Bank)	Bank		
Previous 2 NRC Exams (ILT Only)	No		
Cognitive Level	**Memory or Fundamental Knowledge**	**Comprehension or Analysis**	X

EXAMINATION ANSWER KEY

2016 RO NRC TEST

10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	263000	**PRA:**	No	
Safety Function(s):	6	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

21 **ID: 1248385** **Points: 1.00**

The plant is in cold shutdown with the following conditions:

- Shutdown Cooling (SDC) Loops A and B are operating
- SDC Loop C is shutdown
- Recirc Pumps D and E are operating
- Rx Water is 160" and steady
- Recirc Loop Suction Temperatures are 100°F and stable
- It is desired to maintain the plant in cold shutdown

The following events then occurred:

- 'E' Recirc pump tripped and immediate actions of ABN-2 Recirculation System Failures have been taken
- An inadvertent SDC isolation signal caused a loss of SDC.
- The isolation signal has cleared.

What action is required NEXT?

Enter ABN-3, Loss of Shutdown Cooling, THEN...

A. Raise RPV water level to greater than 185 in TAF to establish circulation flow through the steam separators.

B. Restore SDC in accordance with Procedure 305, Shutdown Cooling System Operation.

C. Establish alternate shutdown cooling using Core Spray and EMRVs.

D. Isolate 'E' recirculation loop

Answer: B

Answer Explanation			
K&A	205000 - Shutdown Cooling System (RHR Shutdown Cooling Mode) 2.4.9 - Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (3.8/4.2)		
Level: RO	**Tier:** 2		**Group:** 1
References	ABN-3		

Explanation	Proposed Answer: B Explanation: SDC is on the 'E" Recirculation loop therefore since the 'E' Recirc pump has tripped and been placed in an idle condition and 'D' recirc pump is still running then ABN-3 states that if a loss of SDC occurs due to an inadvertent isolation and the isolation signal is no longer present, then restore SDC per Procedure 305. A. Plausible – This would be the appropriate action per ABN-3 if both recirc pumps tripped concurrently with the loss of SDC and the applicant believes that since SDC is currently tripped that raising level would be appropriate. Since there is still one recirc pump running this action is not required. C. Plausible in that applicant does not recall the immediate actions of ABN-2 to place the 'E' recirc pump in an idled condition then establishing Alternate shutdown cooing systems is apporiate because the 'E" RCP must be in an idled or isolated condition to restore SDC per 305 otherwise short cycling will occur and the Reactor will no be cooled down. Since the immediate actions were taken in ABN-2 which includes idling the 'E' Recirc loop and the SDC isolation signal has been cleared then this is not the correct NEXT step. D. Plausible is the applicant believes that since the 'E' Recirc pump is not running then the pump must be placed in an isolated condition to restore SDC since in an idled condition the discharge bypass valve is still open and short cycling SDC will still occur. The SDC procedure requires the 'E' recirc pump to either be running or in an idle condition to prevent short cyclingtherefore isolating the loop is not required.			
Lesson Plan **Learning Objective/**	2621.828.0.0045 - SHUTDOWN COOLING SYSTEM SDC-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, EOP & EOP support procedures and EP Procedures.			
References Provided	**ILT:** None			**LORT:** Open
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	10	**55.43b**	
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	205000	**PRA:**	No	
Safety Function(s):	4	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

22 **ID: 1248386** **Points: 1.00**

The plant is performing a refuel outage when the following occurred:

- MN BRKR 1C TRIP annunciator alarmed
- MN BRKR 1C 86 LKOUT TRIP annunciator alarmed

Which one of the following describes the response of EDG 1 and the required operator action?

	EDG 1...	Required Operator Action
A.	starts.	Manually close EDG 1 output breaker.
B.	starts.	Verify EDG 1 output breaker automatically closed.
C.	remains in standby.	Manually start EDG 1
D.	remains in standby.	Verify 4160V Bus Tie Breaker EC is tripped.

Answer: D

Answer Explanation				
K&A	264000 - Emergency Generators (Diesel/Jet) 2.4.50 - Ability to verify system alarm setpoints and operate controls identified in the alarm response manual. (4.2/4.0)			
Level: RO	**Tier:** 2		**Group:** 1	
References	RAP-T2A	RAP-T1A		
Explanation	Proposed Answer: D Explanation: Per RAP-T2A, Lockout of Bus 1C will prevent the fast start of Emergency Diesel Generator #1 and diesel generator breaker closure on faulted Bus 1C. The EDG should remain in standby. The action to verify the breaker tripped is directed in RAP-T2A. A. Plausible – Undervoltage on Bus 1C will cause EDG 1 to start. However, the lockout of Bus 1C will prevent the fast start of EDG 1. B. Plausible – This would be the normal response if the lockout had not occurred. C. Plausible – EDG 1 would remain in standby, however it would not be manually started. Plausible if the applicant doesn't understand the reasoning for the EDG start prevention.			
Lesson Plan **Learning Objective/**	2621.828.0.0016 - ELECTRICAL DISTRIBUTION ACD-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, EOP and EOP support procedures, and EP Procedures.			
References Provided	**ILT:** None		**LORT:** Open	

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	10	**55.43b**	
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	264000	**PRA:**	No	
Safety Function(s):	6	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

23 **ID: 1248387** **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 (if any) 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			
K&A	215005 - Average Power Range Monitor/Local Power Range Monitor K6.03 - Knowledge of the effect that a loss or malfunction of the following will have on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM : Detectors (3.1/3.3)		
Level: RO	**Tier:** 2		**Group:** 1
References	NF-AB-770	403	
Explanation	Proposed Answer: B 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. A. Plausible if the applicant believes neutron count rate will affect core thermal power calculations. C. 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. D. 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.		
Lesson Plan Learning Objective/	2621.828.0.0029 - NUCLEAR INSTRUMENTATION NIS-10435 - Given plant operating conditions, describe or explain the purpose(s)/function(s) of the system and its components.		
References Provided	**ILT:** None		**LORT:** Open
Question Source (New, Modified, Bank)	Bank		

Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	215005	**PRA:**	No		
Safety Function(s):	7	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

24 **ID: 1248388** **Points: 1.00**

The plant was at rated power when a small RPV coolant leak resulted in:

- Primary Containment pressure rising and stabilizing at 2.25 psig
- Primary Containment temperature rising and stabilizing at 155°F

The Operator has initiated venting the Torus (through V-28-18 and V-28-47) IAW Support Procedure 31, Venting the Primary Containment to Maintain Pressure Below 3.0 PSIG, with Standby Gas Treatment System 1 (SGTS 1).

5 minutes later, Drywell pressure rose causing an automatic scram.

Which of the following states the impact on the venting process and on SGTS 2?

	Torus Venting	SGTS 2
A.	Vent path isolates	SGTS 2 remains in Standby
B.	Vent path isolates	SGTS 2 immediately starts
C.	Vent path remains open	SGTS 2 remains in Standby
D.	Vent path remains open	SGTS 2 immediately starts

Answer: B

Answer Explanation			
K&A	212000 - Reactor Protection System K1.13 - Knowledge of the physical connections and/or cause-effect relationships between REACTOR PROTECTION SYSTEM and the following: Containment pressure (3.5/3.6)		
Level: RO	**Tier:** 2		**Group:** 1
References	SP-31	SP-1	330
Explanation	Proposed Answer: B Explanation: When DW pressure rises to the scram setpoint (3 psig), the primary containment will isolate which is fed from RPS logic and isolate the Torus vent valves and both SGTS trains receive a start signal. SGTS1 will remain running and SGTS 2 will auto start. A. Plausible – The vent path will isolate. Plausible if the applicant believes an auto start signal is not processed since a SGTS train is already in operation. C. Plausible – If the applicant doesn't know what is isolated on a containment isolation signal and an auto start signal is not processed since a SGTS train is already in operation.. D. Plausible – If the applicant doesn't know what is isolated on a containment isolation signal.		
Lesson Plan Learning Objective/	2621.828.0.0042 - Secondary Containment and SGTS SGT-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.		
References Provided	**ILT:** None		**LORT:** Open

Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	212000	**PRA:**	No	
Safety Function(s):	7	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

25 **ID: 1248389** **Points: 1.00**

The plant is operating at rated conditions with the following:

- RBCCW liquid process radiation monitor indicates 3200 cps.
- No chemicals are being added to the RBCCW system.

Which one of the following describes the plant response and the next required operator action in accordance with ABN-19, RBCCW Failure Response?

	Plant Response	Required Operator Action
A.	A RBCCW High Radiation Alarm is received, **ONLY**.	Isolate makeup to the surge tank.
B.		

	A RBCCW High Radiation Alarm is received, **ONLY**.	Trip any operating RBCCW Pumps.
C.	A RBCCW High Radiation Alarm is received **AND** a RBCCW Drywell Isolation occurs.	Isolate makeup to the surge tank.
D.	A RBCCW High Radiation Alarm is received **AND** a RBCCW Drywell Isolation occurs.	Trip any operating RBCCW Pumps.

Answer: A

Answer Explanation				
K&A	400000 - Component Cooling Water System A2.04 - Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Radiation monitoring system alarm (2.9/3.0)			
Level: RO	**Tier:** 2		**Group:** 1	
References	ABN-19	RAP-10F3F		
Explanation	Proposed Answer: A Explanation: There are no automatic actions associated with the radiation monitoring detectors in the RBCCW system. Therefore the operators are alerted to the situation by alarm only and must carry out the actions specified in the RAP. Since high radiation conditions in RBCCW are caused by leakage into the system, The flowchart, ABN-19-3 is used which directs the operator to isolate makeup to the surge tank. B. Plausible – Only a radiation alarm is received. Tripping the RBCCW pumps would minimize any radioactive release. However, this is not directed by ABN-19. C. Plausible – Isolating the surge tank is the correct action. However, no automatic isolation occurs. D. Plausible – Tripping the RBCCW pumps would minimize any radioactive release. However, this is not directed by ABN-19.			
Lesson Plan Learning Objective/	2621.828.0.0035 - RBCCW RBC-00048 - List possible causes, system response and affected RBCCW components for an auto isolation signal.			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	

10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	400000	**PRA:**	No	
Safety Function(s):	8	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

26 **ID: 1248390** **Points: 1.00**

Given the following conditions:

The plant is at 71% power and steady.
Reactor pressure is 1019 psig.
Feedwater level transmitter selector has ONLY the "A" light illuminated.

BOP operator reports
"A" GEMAC indicates 163" and rising.
"B" and "C" GEMACs are tracking together but not corresponding with "A" GEMAC.

With NO operator action, the actual reactor water level will _____(1)_______ and the controlling level instrument is ______(2)______.

	1	2
A.	lower	the "A" [ID13A] level transmitter
B.	rise	the "A" [ID13A] level transmitter
C.	remain constant	the "C" (ID13C) level transmitter
D.	remain constant	the "B" (ID13B) level transmitter

Answer: A

Answer Explanation			
K&A	259002 - Reactor Water Level Control System K3.01 - Knowledge of the effect that a loss or malfunction of the REACTOR WATER LEVEL CONTROL SYSTEM will have on following: Reactor water level (3.8/3.8)		
Level: RO	**Tier:** 2	**Group:** 1	
References	MDD-OC-625-B DIV I	RAP-J9c	
Explanation	Proposed Answer: A Explanation: Question indicates a failure of the A GEMAC transmitter only with no leaks (the shared instrument on that reference leg is C GEMAC, which is not corresponding with A GEMAC). With the MLC in 'A" as indicated by only one light on the selector, the controlling instrument will not switch to the B GEMAC and actual Rx water level will lower due to indicated level rising. B. Plausible since level will lower and the controlling instrument will be the "A" due to being selected. It is plausible if the operator believes that since A is rising it will send a raise reactor level and maintain control.B. C. Plausible since level will lower and the controlling instrument will be the "A" due to being selected. It is plausible if the operator believes that it will automatically switch to C controller. D. Plausible since level will lower and the controlling instrument will be the "A" due to being selected. It is plausible if the operator believes that it will automatically switch to B controller.		
Lesson Plan Learning Objective/	2621.828.0.0018 - FEEDWATER CONTROL SYSTEM FWC-10444 - Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

References Provided	**ILT:** None			**LORT:** Open	
Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X	
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	259002	**PRA:**	No		
Safety Function(s):	2	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

27 **ID: 1248391** **Points: 1.00**

Given the following conditions:

- The plant is operating at 100% power
- The main generator is carrying 100 MVARs

An event occurred that caused a complete loss of AC power to the Automatic Voltage Regulator (AVR).

Which one of the following conditions occurs as a result of this event? (if Any)

A. Main Generator Trip

B. Main Generator runback

C. Swaps to Main Generator manual voltage regulation

D. Remains in Main Generator automatic voltage regulation

Answer: A

Answer Explanation			
K&A	245000 - Main Turbine Generator and Auxiliary Systems K1.01 - Knowledge of the physical connections and/or cause-effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: A. C. electrical distribution (3.2/3.3)		
Level: RO	**Tier:** 2		**Group:** 2
References	ABN-10	ABN-44	
Explanation	Proposed Answer: A Explanation: Loss of power to the AVR will cause a turbine-generator trip due to loss of main generator field. B. Plausible – if the applicant believes that a runback would occur with a loss of the AVR but a runback would occur if Stator cooing was lost not the AVR. C. Plausible – if the applicant believes that since the AVR lost power generator voltage would swap to manual as it would if there was an issue with the AVR but still had power to it but since it has lost power would cause a turbine-generator trip instead. D. Plausible - if the applicant believes that the generator would remain in automatic regulation because the AVR backup power supply is DC then the AVR would still be available. The AVR does have a backup power supply supplied by an ATS switch but the backup power supply is AC power as well and the stem states that only AC power was lost therefore it would create a trip on a loss of power.		
Lesson Plan **Learning Objective/**	2621.828.0.0025 - MAIN GENERATOR GEN-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status, GEN-10446 - Identify and explain system operating controls/indications under all plant operating conditions.		
References Provided	**ILT:** None		**LORT:** Open

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Question Source (New, Modified, Bank)	Modified				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	x	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	245000	**PRA:**	No		
Safety Function(s):	4	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

28 **ID: 1248392** **Points: 1.00**

The plant is operating at 50% power with the following:

- Feedwater Pumps 1A and 1B are operating.
- Condensate Pumps 1A and 1B are operating.
- The remaining Condensate and Feedwater pumps are in off.

An electrical fault causes bus 1A to de-energize.

Which one of the following describes the impact on the Feedwater Pumps with no operator action?

A. Feedwater Pump 1C loses power and Feedwater pump 1B de-energizes.

B. Feedwater Pump 1C loses power and Feedwater pump 1B remains running.

C. Feedwater Pump 1A loses power and Feedwater pump 1B de-energizes.

D. Feedwater Pump 1A loses power and Feedwater pump 1B remains running.

Answer: D

<table>
<tr><th colspan="5">Answer Explanation</th></tr>
<tr><td>K&A</td><td colspan="4">259001 - Reactor Feedwater System
K2.01 - Knowledge of electrical power supplies to the following: Reactor feedwater pump(s): Motor-Driven-Only (3.3/3.3)</td></tr>
<tr><td>Level: RO</td><td colspan="3">Tier: 2</td><td>Group: 2</td></tr>
<tr><td>References</td><td>316</td><td>317</td><td colspan="2">RAP-S2e</td></tr>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: D

Explanation: Feedwater Pumps 1-B and 1-C receive power from 4160 VAC bus 1B, and pump 1-A is powered from bus 1A. Therefore on a loss of bus 1A, FW Pump 1-A would de-energize and lose power. FW Pump 1-C would be available, however with no operator actions 1C would remain off.

A. Plausible since two FW pumps are powered from one 4160V bus and one FW pump is powered from a different 4160V bus. The applicant needs to recognize that FW Pump 1-A is the only FW Pump affected by the power loss.
B. Plausible since two FW pumps are powered from one 4160V bus and one FW pump is powered from a different 4160V bus. The applicant needs to recognize that FW Pump 1-A is the only FW Pump affected by the power loss.
C. Plausible since FW Pump 1-A is de-energized with the power loss. The applicant need to recognize that FW Pump 1-A is the only FW Pump affected by the power loss.</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="4">2621.828.0.0017 – Fed & Condensate System
CFW-10453 - Explain or describe how this system is interrelated with other plant systems.</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">New</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	259001	**PRA:**	No	
Safety Function(s):	2	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

29 **ID: 1248393** **Points: 1.00**

The plant was at rated power when a LOCA occurred. Containment Spray Pumps 51B and 51C have been started in the Drywell Spray Mode. The following annunciator then alarmed:

- VITAL POWER DC PWR LOST - BUS C UV

The Operator reports 0 volts on DC Bus C.

Which of the following states the impact on the running Containment Spray Systems?

	SYSTEM 1	SYSTEM 2
A.	Swaps to Torus Cooling mode	Swaps to Torus Cooling mode
B.	Swaps to Torus Cooling mode	Remains in Drywell Spray mode
C.	Remains in Drywell Spray mode	Swaps to Torus Cooling mode
D.	Remains in Drywell Spray mode	Remains in Drywell Spray mode

Answer: B

Answer Explanation			
K&A	226001 - RHR/LPCI: Containment Spray System Mode K3.03 - Knowledge of the effect that a loss or malfunction of the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE will have on following: Containment/drywell/suppression chamber components, continued operation with elevated pressure and/or temperature and/or level (2.9/3.2)		
Level: RO	**Tier:** 2		**Group:** 2
References	ABN-55	GE 237E901 sh. 1	GE 116B8328 sh. 11

EXAMINATION ANSWER KEY

2016 RO NRC TEST

<table>
<tr><td rowspan="1">Explanation</td><td colspan="5">Proposed Answer: B

Explanation: Oyster Creek does not have an RHR/LPCI system. The equivalent for this KA is the containment spray system. A LOCA is in progress causing elevated containment parameters, such that the containment spray system is requird to spray the DW in order to control those parameters. When Vital DC bus C power is lost, this represents the loss or malfunction to the containmen spray system due to the loss of control power for system #1 (Pump 51B. The question is testing knowledge of the effect on system components and their ability to continue to control DW parameters, since the applicant must determine that a loss of DC control power to system #1 will cause that system to transfer to Torus cooling mode, and will no longer be available to control the elevated drywell parameters.The DC control logic for DW Sprays System 1 is provided from DC-F (fed from DC-C and has no alternate power supply), and DC-D (fed from DC-B) provides DC control power for System 2. When in the DW spray mode and the DC control power is lost, the affected system converts to the Torus Cooling mode and the pumps remain running. Pump 51B is in System 1 and 51C is in System 2. Therefore, when the System 1 DC control power is lost, System 1 will convert to torus cooling. System 2 is not affected by the DC loss and Pump 51C remains in DW spray mode.

A. Plausible – System 1 will swap to Torus Cooling Mode. However, System 2 remains unaffected.
C. Plausible – This would be the impact of a loss of the system 2 DC control power Bus.
D. Plausible – This would be the impact of a loss of DC "B" and system 2 DC control power auto swapped to backup power supply.</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="5">2621.828.0.0009 - CONTAINMENT SPRAY/ESW SYSTEMS
CNS-10449 - State the function and interpretation of system alarms, alone and in combination, as applicable in accordance with the system RAPS.</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td colspan="2">LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="5">Bank</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="5">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td>X</td><td>Comprehension or Analysis</td><td colspan="2"></td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>7</td><td>55.43b</td><td colspan="2"></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="5">Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features</td></tr>
<tr><td>Justification for LORT K&A <3.0</td><td colspan="5">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="5">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="5">1</td></tr>
<tr><td>System ID No.:</td><td>226001</td><td>PRA:</td><td>No</td><td colspan="2"></td></tr>
</table>

Safety Function(s):	5	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

30 **ID: 1248394** **Points: 1.00**

A reactor startup is in progress. Control Rod 34-51 is being withdrawn to position 48.
Upon reaching position 48 the following annunciator came into alarm:

- ROD OVERTRAVEL

Which of the following indications on Panel 4F would confirm Control Rod 34-51 is uncoupled?

A. Black backlight with "48"

B. Black backlight and no numbers

C. Red backlight with "48"

D. Red backlight and no numbers

Answer: B

Answer Explanation			
K&A	201003 - Control Rod and Drive Mechanism K4.05 - Knowledge of CONTROL ROD AND DRIVE MECHANISM design feature(s) and/or interlocks which provide for the following: Rod position indication (3.2/3.3)		
Level: RO	**Tier:** 2		**Group:** 2
References	302.2	RAP-Ha	
Explanation	Proposed Answer: B Explanation: IAW 302.2, Control Rod Drive Manual Control System, if a control rod became uncoupled, the rod position display (on Panel 4F) will go dark (black with no position indication) and the ROD OVERTRAVEL alarm (H5a) will annunciate. These design features are what the applicant will use to detect if an uncoupled control rod condition exists. A. Plausible since these would be control rod display indications under conditions other than an uncoupled rod. C. Plausible since these would be control rod display indications under conditions other than an uncoupled rod. D. Plausible since these would be control rod display indications under conditions other than an uncoupled rod.		
Lesson Plan **Learning Objective/**	2621.828.0.0011 - CONTROL ROD DRIVE AND HYDRAULICS CRD-10460 - Describe the CRDM design features and/or interlocks which provide for the following: Detection of an uncoupled control rod, Slowing the drive mechanism near the end of travel following a scram, The use of either the accumulator or reactor water to scram the control rod, Maintaining the control rod at a given location.		
References Provided	**ILT:** None		**LORT:** Open
Question Source (New, Modified, Bank)	Bank		
Previous 2 NRC Exams (ILT Only)	No		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b	7	55.43b	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	201003	PRA:	No	
Safety Function(s):	1	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

31 **ID: 1248395** **Points: 1.00**

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

- Rod select power switch (4F) is in the OFF Position
- Control rod 26-27 is at position 48
- Then, control rod 26-27 reed switch S47 (for position 47) fails in the closed position
- No Control rod movement occurs

Which one of the following describes the impact of this switch failure?

RAP-H6a, CONTROL ROD DRIFT, will annunciate...

A. at this time.

B. only when Rod Select Power is turned on.

C. only when Control Rod 26-27 is selected for movement.

D. only when Control Rod 26-27 is selected AND the PLC timer has started.

Answer: A

Answer Explanation			
K&A	214000 - Rod Position Information System K5.01 - Knowledge of the operational implications of the following concepts as they apply to ROD POSITION INFORMATION SYSTEM : Reed switches (2.7/2.8)		
Level: RO	**Tier:** 2		**Group:** 2
References	RAP-H6a		
Explanation	Proposed Answer: A Explanation: The rod drift alarm will be received due to an odd reed switch being closed with the rod not selected for movement. B. Plausible if the applicant thinks the rod selection circuit must have power in order to process a rod drift. The rod select power switch provides power to the entire rod selection circuit. C. Plausible since control rod selection is an input to the rod drift circuitry. However, the rod must NOT be selected to process the rod drift alarm. If a different rod were selected, the control rod drift annunciator would alarm. D. Plausible if the applicant believes the PLC timer is an input to the rod drift circuitry, as is the case at other facilities. However, at Oyster Creek, the PLC timer is not an input to the rod drift alarm.		
Lesson Plan Learning Objective/	N-OC-2621.828.0.0036 - REACTOR MANUAL CONTROL SYSTEMS RMC-10446- Identify and explain system operating indications under all plant operating conditions.		
References Provided	**ILT:** None		**LORT:** Open
Question Source (New, Modified, Bank)	New		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	5	**55.43b**	
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of lead changes, and operating limitations and reasons for these operating characteristics			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	214000	**PRA:**	No	
Safety Function(s):	7	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

32 **ID: 1248396** **Points: 1.00**

A core shuffle is in progress with all control rods fully inserted. The following two moves have been made.

- A double blade guide (DBG) was lifted in the Spent Fuel Pool. All indications on the Refuel Bridge were correct for lifting the DBG. The DBG was transferred and released in the core.
- A fuel bundle was lifted in the Spent Fuel Pool. All indications on the Refuel Bridge were correct for lifting the fuel bundle. The fuel bundle was transferred over the core.

During both moves, the following indications were observed during the entire time of the moves:

- In the Control Room, the REFUEL INTERLOCK light on Panel 4F is NOT illuminated.
- On the Refuel Bridge, the ROD BLOCK INTERLOCK light is NOT illuminated.

Which one of the following describes the refueling interlock/limit switch that has failed and when the rod block should have been in effect?

	Interlock/Limit Switch Failure	Rod Block in Effect
A.	The bridge reverse stop interlock	When the fuel bundle was over the core.
B.	The bridge reverse stop interlock	When the double blade guide was over the core.
C.	The "platform over core" limit switches (LS-1 / LS-2)	When the fuel bundle was over the core.
D.	The "platform over core" limit switches (LS-1 / LS-2)	When the double blade guide was over the core.

Answer: C

Answer Explanation			
K&A	234000 - Fuel Handling K5.02 -Knowledge of the operational implications of the following concepts as they apply to fuel handing equipment: Fuel handling equipment interlocks. (3.1/3.7)		
Level: RO	**Tier:** 2		**Group:** 2
References	205	656.4.001	RAP-H7a

EXAMINATION ANSWER KEY

2016 RO NRC TEST

<table>
<tr><td rowspan="1">Explanation</td><td colspan="4">Proposed Answer: C

Explanation: The over-the-core limit switch has failed which prevents a RMCS rod out block from being generated when the bridge is loaded and over the reactor core. The failure can only be recognized after the hoist has fuel loaded and the refuel bridge is over the core. At that point the REFUEL INTERLOCK indicator light on Panel 4F and the ROD BLOCK INTERLOCK Light on the refueling bridge will light and the ROD BLOCK annunciator should alarm. A double blade guide weighs approximately 180 lbs. This is significantly less than the HOIST LOADED setpoint of 485 lbs which inputs to the refueling interlocks.

A. Plausible for those candidates that do not recognize that even if the rod out interlock had failed, with the “over core” limit switch functional, the ROD BLOCK and other normal indications would be received. The refuel bridge reverse stop interlock would generate a bridge reverse stop, fuel hoist interlock, and rod block would be received, not just a rod block.
B. Plausible for those candidates that do not recognize that even if the rod out interlock had failed, with the “over core” limit switch functional the ROD BLOCK and other normal indications would be received. The refuel bridge reverse stop interlock would generate a bridge reverse stop, fuel hoist interlock, and rod block would be received, not just a rod block. A failure of this interlock would only be recognized with a fuel bundle loaded on the hoist since the double blade guide does not weigh enough to meet the hoist loading requirement to complete the refuel interlock.
D. Plausible for those candidates that are unsure of the double blade guide weight or unaware of the weight requirement in the refuel interlock. The “over core” limit switch has failed; however the double blade guide does not weigh enough to meet the hoist loading requirement to complete the refuel interlock and create the Rod block.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.812.0.0003- Refueling
RFL-00325 - Given Procedure 656.4.001, Refueling Bridge Interlock Circuit surveillance, explain the purpose of each step/section of the procedure and the expected system response to these steps.</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">New</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>7</td><td>55.43b</td><td></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features</td></tr>
<tr><td>Justification Justification for LORT K&A <3.0</td><td colspan="4">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="4">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="4">1</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

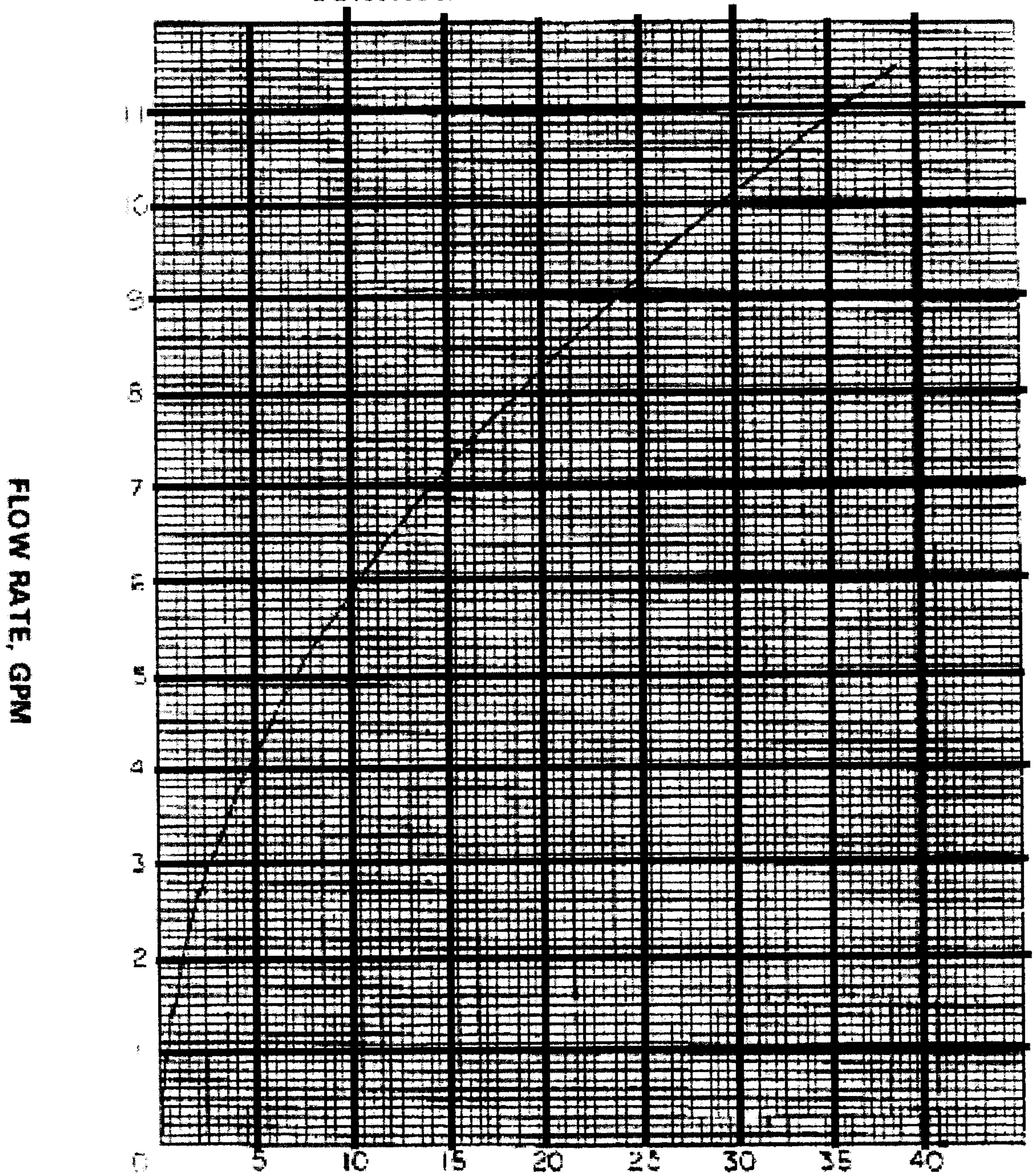
System ID No.:	234000	**PRA:**	No	
Safety Function(s):	8	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

33 **ID: 1248397** **Points: 1.00**

The plant was at rated power with surveillance procedure 617.4.002, CRD Exercise and Flow Test/IST Cooling Water Header Check Valve, in-progress.

If the procedure requires an IR generated for any stall flow > 5 GPM for a rod withdraw, which of the following represents the **SMALLEST** CRD drive flow ΔP which requires an IR generated?
(SEE ATTACHED)

CONTROL ROD DRIVE FLOW



CONTROL ROD DRIVE FLOW
ΔP, PSI

A. 3 ΔP psi

B. 5 ΔP psi

C. 6 ΔP psi

D. 8 ΔP psi

Answer: D

Answer Explanation			
K&A	201002 - Reactor Manual Control System A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the REACTOR MANUAL CONTROL SYSTEM controls including: CRD drive water flow (2.8/2.8)		
Level: RO	**Tier:** 2		**Group:** 2
References	617.4.002		
Explanation	Proposed Answer: D Explanation: A flow of 5 gpm corresponds to approximately 7.5 psid. 8 psid is the smallest differential pressure listed that would exceed the 5 gpm stall flow limit. NOTE: Question matches KA since it demonstrates the ability to monitor CRD drive flow during a control rod withdraw. A. Plausible if the applicant confuses axes on the graph. 5 psid equates to between 4 gpm and 5 gpm. B. Plausible if the applicant confuses axes on the graph. 5 psid equates to between 4 gpm and 5 gpm. C. Plausible since a flow of 5 gpm corresponds to approximately 7.5 psid. The 7 psid distractor represents the largest ΔP listed prior to exceeding 5 gpm stall flow.		
Lesson Plan Learning Objective/	2621.828.0.0036 - REACTOR MANUAL CONTROL SYSTEMS 217-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.		
References Provided	**ILT:** None		**LORT:** Open
Question Source (New, Modified, Bank)	Bank		
Previous 2 NRC Exams (ILT Only)	No		
Cognitive Level	**Memory or Fundamental Knowledge**	**Comprehension or Analysis**	X
10CFR55 Content	**55.41b** 7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features		
Justification for LORT K&A <3.0	N/A		
Time to Complete:	1-2 minutes		
Point Value:	1		
System ID No.:	201002	**PRA:** No	

Safety Function(s):	1	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

34 **ID: 1248398** **Points: 1.00**

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

- Four (4) recirculation loops are in service
- Recirculation loop D is IDLE

Then, the following annunciators are received

- RAP-T1C, MN BKR 1B TRIP
- RAP-T2C, MN BKR 1B 86 LKOUT TRIP

Which one of the following describes the effect on recirculation pumps and the actions required in accordance with ABN-2, Recirculation System Failures?

	Effect on Recirculation Pumps	Actions Required
A.	Only Pump B trips	Close the discharge valve for Pump B
B.	Only Pumps A, C, and E trip	Close the discharge valves for Pumps A, C, and E
C.	Only Pump B trips	Scram the reactor due to reduced Recirc flow
D.	Only Pumps A, C, and E trip	Scram the reactor due to reduced Recirc flow

Answer: A

Answer Explanation			
K&A	202001 - Recirculation System A2.19 - Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of A.C. power: Plant-Specific (3.1/3.2)		
Level: RO	**Tier:** 2		**Group:** 2
References	ABN-2,	301.2	
Explanation	Proposed Answer: A Explanation: A loss of Bus 1 B causes a loss of power to recirc pumps B and D. With four loops in operation and recirc loop D IDLE, the power board loss will result in the trip of only pump B. Three recirc loops will remain in operation. ABN-2 directs closing the discharge valve on the tripped pumps. B. Plausible since these would be the pumps to trip on a loss of Bus 1A vice 1B. ABN-2 directs closing the discharge valve on the tripped pumps. C. Plausible since pump B is the only tripped pump. Additionally, this would be correct id recirc loop D was in service vice idle. ABN-2 directs inserting a manual scram if < 3 recirc loops are operating OR if multiple recirc pump trips have occurred. D. Plausible since this would be the correct answer for a loss of Bus 1A vice 1B.		

Lesson Plan **Learning Objective/**	2621.828.0.0038 - REACTOR RECIRCULATION SYSTEM RRS-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, EOP & EOP support procedures and EP procedures.			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	Modified (RR-14a)			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	5	**55.43b**	
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of lead changes, and operating limitations and reasons for these operating characteristics			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	201002	**PRA:**	No	
Safety Function(s):	1	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

35 **ID: 1248399** **Points: 1.00**

A radiological release is in progress with the following:

- Control Room Ventilation has been placed in partial recirculation A/C mode.
- Control Room air temperature is 75°F
- Outside air temperature is 65°F

Then, a loss of offsite power occurs.

Which one of the following describes the status of control room ventilation and the allowable operator actions in accordance with Procedure 331.1, Control Room and Old Cable Spreading Room Heating, Ventilation and Air Conditioning System?

	Control Room Ventilation...	Allowable action
A.	automatically shuts down.	Align Control Room ventilation to purge A/C mode.
B.	automatically shuts down.	Operate Control Room ventilation in FANS-ONLY mode.
C.	remains running in partial recirculation A/C mode.	Align Control Room ventilation to purge A/C mode.
D.	remains running in partial recirculation A/C mode.	Operate Control Room ventilation in FANS-ONLY mode.

Answer: B

Answer Explanation			
K&A	290003 - Control Room Heating, Ventilation and Air Conditioning A3.01 - Ability to monitor automatic operations of the CONTROL ROOM HVAC including: Initiation/reconfiguration (3.3/3.5)		
Level: RO	**Tier:** 2		**Group:** 2
References	331.1		

EXAMINATION ANSWER KEY

<table>
<tr><td rowspan="1">Explanation</td><td colspan="4">Proposed Answer: B

Explanation: Procedure 331.1, section 8.1 discusses impacts on control room ventilation for a loss of offsite power. A Note at step 8.1.2.3 states, "As a result of a loss of offsite power, the Control Room HVAC system will automatically shut down... Do not run in FANS-ONLY mode if Control Room temperature is less than outdoor temperature." Step 8.1.22 says to only operate fans, not compressors during a loss of offsite power.

A. Plausible since the control room ventilation does automatically shutdown. The note at step 8.1.2.3 does allow the use of purge mode, however it must be fans only, not A/C
C. Plausible if the applicant doesn't know a loss of offsite power causes and automatic shutdown of control room ventilation. The note at step 8.1.2.3 does allow the use of purge mode, however it must be fans only, not A/C
D. Plausible if the applicant doesn't know a loss of offsite power causes and automatic shutdown of control room ventilation. Operating in FANS-ONLY mode is allowed.</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="4">2621.828.0.0054 - Turbine Building And Misc. Ventilation Systems TMV-10444 - Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">New</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>7</td><td>55.43b</td><td></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features</td></tr>
<tr><td>Justification for LORT K&A <3.0</td><td colspan="4">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="4">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="4">1</td></tr>
<tr><td>System ID No.:</td><td>290003</td><td>PRA:</td><td>No</td><td></td></tr>
<tr><td>Safety Function(s):</td><td>9</td><td>☒ ILT</td><td></td><td></td></tr>
<tr><td>Category(s) (LORT Only):</td><td>N/A</td><td>☐ LORT</td><td></td><td></td></tr>
</table>

36 **ID: 1248400** **Points: 1.00**

Given the following plant conditions:

- The plant is operating at 100% power.
- Cleanup Recirc Pump 1B is running.
- A failure in the air line to the Reactor Cleanup system pressure control valve, PCV-ND11 causes a complete loss of air to that valve.

System pressure downstream of PCV-ND11 __(1)__ and the reactor water cleanup system __(2)__.

	(1)	(2)
A.	rises	trips
B.	rises	does NOT trip
C.	lowers	trips
D.	lowers	does NOT trip

Answer: C

Answer Explanation			
K&A	204000 - Reactor Water Cleanup System A4.05 - Ability to manually operate and/or monitor in the control room: System pressure (2.9/2.8)		
Level: RO	**Tier:** 2	**Group:** 2	
References	FSAR Section 5.4		
Explanation	Proposed Answer: C Explanation: A loss of air to PCV-ND11 causes it to fail closed. The system pressure down stream of this valve will drop as a result. System flow is also affected. A rapid loss of air to ND-11 (<30 sec) will not allow sufficient time for the Hold Pump to recover filter flow before flow drops to < 80 gpm; CU will trip and the running cleanup recirc pump trips. A. Plausible if the applicant doesn't know how the valve will fail on a loss of air. The valve will fail closed causing downstream pressure to lower. The system will trip. B. Plausible if the applicant doesn't know how the valve will fail on a loss of air. The valve will fail closed causing downstream pressure to lower. D. Plausible since downstream system pressure will lower. Also, on a slow loss of air the Filter Hold Pump will be able to recover filter flow before the 80 gpm isolation setpoint is reached.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Lesson Plan **Learning Objective/**	2621.828.0.0039 - REACTOR WATER CLEANUP SYSTEM RCU-10444 - Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.			
References Provided	**ILT:** None			**LORT:** Open
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	204000	**PRA:**	No	
Safety Function(s):	2	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

37 **ID: 1250268** **Points: 1.00**

Given the following plant conditions:

- Refueling is in progress.
- "B" Fuel Pool Cooling Pump (NN-01B) is running.
- "D" Augmented Fuel Pool Cooling Pump (NN-01D) is running.
- Annunciator G-7-a, "SKM SRG TNK LVL LO-LO" is received and acknowledged.

What automatic or manual action occurs or is required?

	NN-01B	**NN-01D**
A.	Must be manually tripped	Must be manually tripped
B.	Will automatically trip	Must be manually tripped
C.	Must be manually tripped	Will automatically trip
D.	Will automatically trip	Will automatically trip

Answer: B

Answer Explanation			
K&A	233000 - FUEL POOL COOLING AND CLEAN-UP 2.4.11 – Knowledge of abnormal conditions procedures (4.0/4.2)		
Level: RO	**Tier:** 2		**Group:** 2
References	ABN-16	RAP G-7-a	
Explanation	Proposed Answer: B Explanation. Upon receipt of the Skimmer Surge Tank Lo-Lo Level, the running fuel pool cooling pump(s) (1A or 1B) will trip. Augmented fuel pool cooling pumps (1C, 1D) trip on low suction pressure but do NOT trip on low skimmer surge tank level. ABN-16 requires manually tripping the Augmented FPC pumps on lo-lo skimmer surge tank level. A is incorrect. NN-01A will trip on lo-lo level. C is incorrect. Exactly opposite of expected plant response. D is incorrect. The Running Augmented Fuel Pool Cooling Pump trips on low suction pressure. Note that the RAP directs the operators to secure this pump upon receipt of this alarm recognizing the low level situation will eventually lead to a low suction trip.		
Lesson Plan Learning Objective/	2621.828.0.0020- FUEL POOL COOLING FPC-10441 - Given the system logic/electrical drawings describe the system trip signals and setpoints and expected system response including power loss or failed components.		
References Provided	**ILT:** None		**LORT:** Open
Question Source (New, Modified, Bank)	Bank		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	10	**55.43b**	
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	233000	**PRA:**	No	
Safety Function(s):	9	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

38 **ID: 1248402** **Points: 1.00**

The plant is at 20% power with Recirc flow at $8.5x10^4$ gpm, when the following conditions occur:

- 9XF-7-b, IP-4B PWR LOST alarm was received
- This condition has existed for 15 minutes.
- Vacuum is 26" and lowering slowly

What is the NEXT action required if IP-4B cannot be restored?

A. Scram the reactor

B. Manually control steam seal pressure

C. Swap pressure reducing valves

D. Reduce recirc flow

Answer: A

Answer Explanation			
K&A	271000 - Offgas System A2.08 - Ability to (a) predict the impacts of the following on the OFFGAS SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. distribution failures (2.5/2.7)		
Level: RO	**Tier:** 2		**Group:** 2
References	ABN-58	ABN-14	

<table>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: A

Explanation: A loss of IP-4B does cause both offgas radiation monitors to de-energize. the applicant must inderstand that when this happens a 15 minute timer starts. After the 15 minute wait, the offgas system isolates and condenser vacuum degrades to the auto scram setpoint. ABN-14 directs the operator to reduce power with recirc not lower than $8.5x10^4$ gpm but the stem already has recirc flow at $8.5x10^4$ gpm therefore a reactor scram is required since vacuum is still lowering slowly and will continue to lower until the automatic scram setpoint is reached.

B. Plausible if the applicant believes that the loss of IP-4B (480V AC bus) trips the automatic controls for the gland steal regulator. If that were the case it is appropriate to take manual control per ABN-14 to restore gland seal if the gland seals are not operating properly to correct vacuum or if the applicant the exhaust blowers trip. But since MCC-1B12A powers the steam seal regulator and MCC-1A13 or MCC-1B13 power the exhauster blowers a loss of IP-4B does not cause a loss of vacuum due to gland seals and since they are operating appropriately this actions is not appropriate for this condition.
C. Plausible if the applicant believes that the loss of IP-4B trips the inservice pressure regulator which is causing vacuum to lower then this would be appropriate per ABN-14 but since the loss of IP-4B does not affect the pressure regulators as it is powered by MCC-1B11a and they are operating as designed this action is not appropriate for these conditions.
D. Plausible if the applicant believes that a power reduction is allowed with the stem conditions then is action is appropriate but since ABN-14 does not allow recirc flow to be lowered to less than $8.5x10^4$ gpm then this action is not appropriate</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.828.0.0002 – Air Extraction and Off-Gas
AEG-00099- Interpret given Control Room and/or local Off-Gas system indications and evaluate them in terms of limits and trends, using available data..</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">New</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>5</td><td>55.43b</td><td></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of lead changes, and operating limitations and reasons for these operating characteristics</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	271000	**PRA:**	No	
Safety Function(s):	9	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

39 **ID: 1248403** **Points: 1.00**

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

RAP C-3-f, DW PRESS HI-LO, annunciates.
Time between pump down of the Drywell floor drain sump has shortened.
Drywell pressure has risen 0.2 psig.
Drywell average temperature has risen 4°F.
Drywell airborne radioactivity levels are unchanged.

Which one of the following conditions is most likely to be causing the observed parameters?

A. Catastrophic failure of both seals on a Recirculation pump.

B. Operating Drywell cooler has a RBCCW leak on the cooler inlet.

C. Operating Drywell cooler has a RBCCW leak on the cooler outlet.

D. EMRV is leaking by with an unseated EMRV tailpiece vacuum breaker.

Answer: B

Answer Explanation			
K&A	295018 - Partial or Complete Loss of Component Cooling Water AK1.01 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Effects on component/system operations (3.5/3.6)		
Level: RO	**Tier:** 2		**Group:** 2
References	RAP-C3f	312.9	ABN-63
Explanation	Proposed Answer: B Explanation: Loss of function of a Drywell cooler, such as would occur if an RBCLC leak develops before the inlet, is expected to cause a rise in Drywell pressure. Procedure 312.9 contains a caution statement which indicates removing fans from service will cause drywell pressure to rise. This would also cause a corresponding rise in Drywell temperature. Drywell airborne radiation levels would not be expected to change since RBCCW water is not contaminated. A. Plausible since failed recirc pump seals would provide indications of rising temperature and pressure. However, airborne radioactivity levels would also be expected to rise. C. Plausible if the applicant doesn't know the functional layout of RBCCW and drywell coolers. Since the leak is on the outlet then flow is unchanged or goes up. Therefore DW pressure and temperature would remain the same or lower, not rise, D. Plausible since a failed open EMRV would provide indications of rising temperature and pressure. However, airborne radioactivity levels would also be expected to rise.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Lesson Plan **Learning Objective/**	2621.828.0.0035 - Reactor Building Closed Cooling Water RBC-00057 – State how Service Water, Shutdown Cooling, Reactor Cleanup, Primary Containment, AC Electrical Distribution and chemical treatment systems interrelate with the RBCCW system			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	8	**55.43b**	
10CFR55 Explanation	Components, capacity, and functions of emergency systems			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295018	**PRA:**	No	
Safety Function(s):	11	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

40 **ID: 1248404** **Points: 1.00**

A loss of coolant accident has resulted in the following:

- The RPV has been depressurized using ICs and EMRVs
- Core Spray is injecting and maintaining Reactor water level
- Containment Sprays have been utilized to lower Containment pressure
- Containment sprays are no longer running
- Torus water temperature is 200°F and stable
- Torus water level is 140 inches and stable
- Torus pressure is 2.9 psig and slowly rising
- Drywell pressure is 3.9 psig and slowly rising
- Core Spray System 2 parallel isolation valves 20-21 and 20-41 have failed closed.
- Core Spray pump NZ01A has tripped due to an electrical fault

Which one of the following states the maximum Core Spray flow (gpm) that may be used for RPV injection while maintaining Core Spray within the NPSH limit?

See attached Graphs

Assume PSTRAINER = 0.5

FIGURE C
CORE AND CONTAINMENT SPRAY STATIC HEAD CURVE

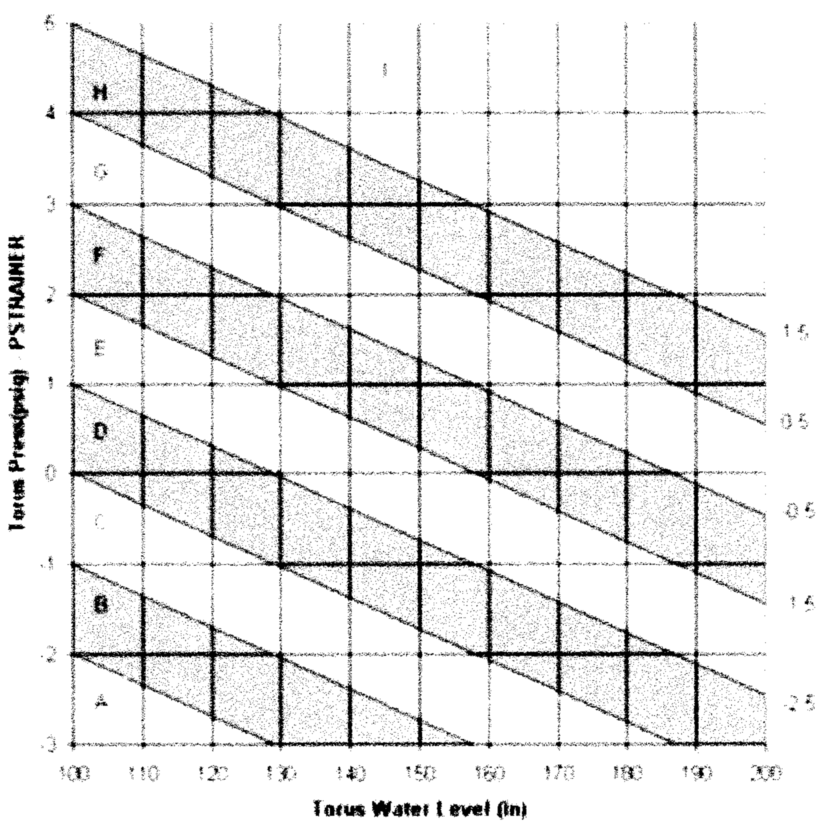
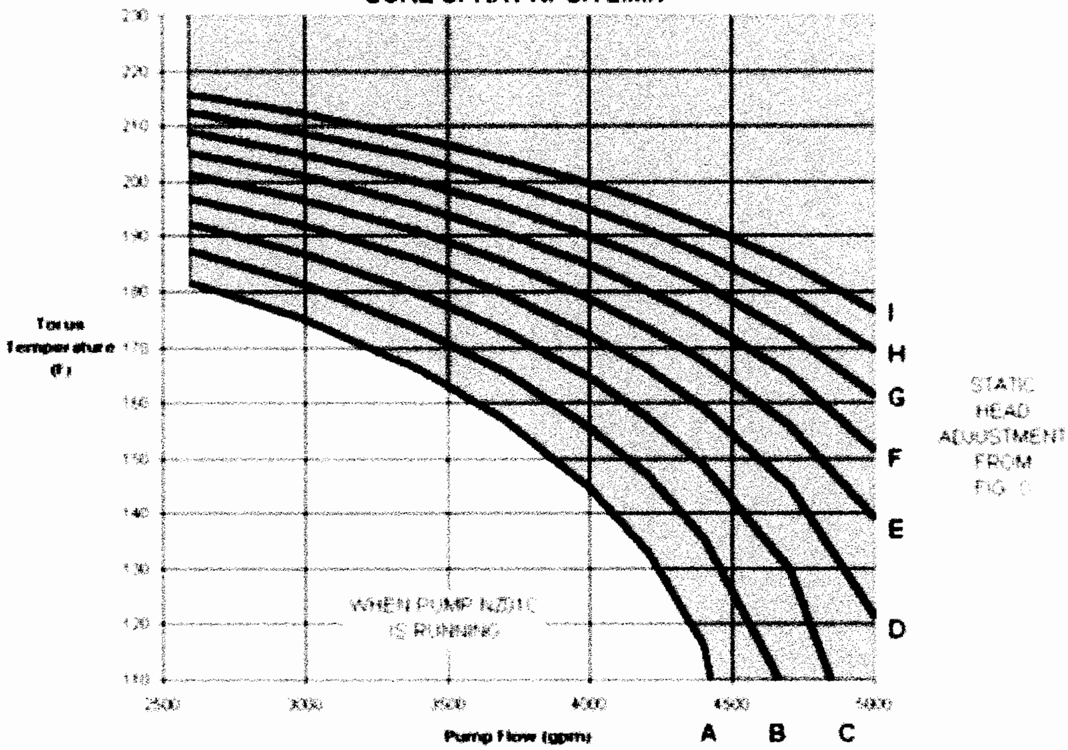


FIGURE B
CORE SPRAY NPSH LIMIT



A. 3200 gpm

B. 3350 gpm

C. 3640 gpm

D. 3750 gpm

Answer: B

Answer Explanation		
K&A	295026 - Suppression Pool High Water Temperature EK1.01 - Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE : Pump NPSH (3.0/.3.4)	
Level: RO	**Tier:** 1	**Group:** 1
References	EMG-SP4	

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Explanation	Proposed Answer: B Explanation: The only available Core Spray pump is a system 1 pump NZ01C. The given containment parameters require the use of curve G, IAW Figure C. Using curve G on figure B, 200F torus temperature intersects curve G at approximately 3400 gpm therefore 3350 gpm is the maximum distractor available. A. Plausible – This would be the correct answer if the applicant uses curve F of figure B. C. Plausible – This represents the rated flow value for system 2 core spray pumps. D. Plausible – If the applicant does not subtract PSTRAINER from torus pressure, then curve H will be used on figure B. This would yield approximately a 3750 gpm limit.			
Lesson Plan **Learning Objective/**	2621.828.0.0010 - CORE SPRAY SYSTEM CSS-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	8	**55.43b**	
10CFR55 Explanation	Components, capacity, and functions of emergency systems			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295026	**PRA:**	No	
Safety Function(s):	10	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

41 **ID: 1248405** **Points: 1.00**

The plant was at rated power when a **STATION BLACKOUT** occurred.

Answer the following questions as they relate to the Isolation Condenser System, with NO AC power available?

1. Can the Isolation Condenser System be manually initiated from the Control Room?
2. Can makeup water be provided to the Isolation Condenser shells (includes both Control Room and in-plant actions)?

	1	2
A.	Yes	No
B.	Yes	Yes
C.	No	No
D.	No	Yes

Answer: B

Answer Explanation			
K&A	295003 - Partial or Complete Loss of A.C. Power AK1.06 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Station blackout: Plant-Specific (3.8/4.0)		
Level: RO	**Tier:** 1		**Group:** 1
References	ABN-35	ABN-36	307

<table>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: B
Explanation: The plant was at power when a station blackout occurred. There is no AC power in the station. In the normal configuration, the steam admission valves to each IC are open, one condensate return valve is open, and the second condensate return valve is closed. The closed valve is DC powered and can be manipulated with a loss of AC power.
Filling of the shells usually requires AC power to a water pump. With AC gone, these AC powered pumps are lost. But the shells can also be filled by the Fire Protection water system, which under the given conditions, will be pressurized by diesel driven fire pumps. The makeup valves are air operated, with air accumulators, and fail closed on loss of air. Even if the accumulators discharged, they can be manually manipulated in the plant locally.
Therefore, the isolation condensers can be initiated in the control room and the shells can be filled from fire protection with the total loss of AC power.

A. Plausible since normal makeup is lost to the IC shells.
C. Plausible if the applicant doesn't know which condensate return valve is normally shut.
D. Plausible if the applicant doesn't know which condensate return valve is normally shut. Also, normal makeup is lost to the IC shells. However fire water is available using the diesel fire pumps</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="4">2621.828.0.0023 - ISOLATION CONDENSERS
ICS-02338 - Given plant conditions, EVALUATE the impact on the Isolation Condenser System and the plant.</td></tr>
<tr><td>References Provided</td><td>ILT: None</td><td></td><td colspan="2">LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">Bank</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>8</td><td>55.43b</td><td></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Components, capacity, and functions of emergency systems</td></tr>
<tr><td>Justification for LORT K&A <3.0</td><td colspan="4">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="4">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="4">1</td></tr>
<tr><td>System ID No.:</td><td>295003</td><td>PRA:</td><td>No</td><td></td></tr>
<tr><td>Safety Function(s):</td><td>11</td><td>☒ ILT</td><td></td><td></td></tr>
<tr><td>Category(s) (LORT Only):</td><td>N/A</td><td>☐ LORT</td><td></td><td></td></tr>
</table>

42 **ID: 1248406** **Points: 1.00**

The plant was at rated power when a LOCA occurred.

Which of the following states the ***sequence*** of automatic RPS protective functions as RPV water level steadily drops from 95" to 82"?

	Occurs First	Occurs Second
A.	**ALL** Recirculation Pumps Trip	Isolation Condensers condensate return valves signaled to open and vent valves to close
B.	Isolation Condensers condensate return valves signaled to open and vent valves to close	**A, B, E ONLY** Recirculation Pumps Trip
C.	Isolation Condensers condensate return valves signaled to open and vent valves to close	**ALL** Recirculation Pumps Trip
D.	**A, B, E ONLY** Recirculation Pumps Trip	Isolation Condensers condensate return valves signaled to open and vent valves to close

Answer: A

Answer Explanation

K&A	295031 - Reactor Low Water Level EK2.11 - Knowledge of the interrelations between REACTOR LOW WATER LEVEL and the following: Reactor protection system (4.4/4.4)		
Level: RO	**Tier:** 1		**Group:** 1
References	RAP-C1a	RAP-C2a	609.3.003
Explanation	Proposed Answer: A Explanation: The isolation condensers auto initiate (after 1.5 seconds) from either a lo-lo RPV water level (90") or RPV high pressure (1051 psig). Recirculation pumps also trip from the same parameters. On lo-lo water level, all recirculation pumps trip immediately. The Lo-Lo- water level comes off of RPS logic relay 1K77 to feed the RCP trips. B. Plausible if the applicant neglects the initiation time delay associated with the ICs. Also, on a high reactor pressure signal, A, B, and E recirculation pumps trip immediately. C and D pumps trip after a 10.5 second time delay. Therefore, it's plausible to say only A, B, and E pumps trip. C. Plausible if the applicant neglects the initiation time delay associated with the ICs. D. Plausible if the applicant mistakes the lo-lo level trip signal for the high reactor pressure trip signal. Choice D would be correct for a high reactor pressure condition (1051 psig).		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Lesson Plan **Learning Objective/**	2621.828.0.0040 - RECIRC FLOW CONTROL RFC-00208 - List and identify the actuating signals and their setpoints for the following Recirc auto trips: Drive Motor Lockout trip, ATWS Recirc Pump trip,Drive Motor breaker trip				
References Provided	**ILT:** None			**LORT:** Open	
Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X	
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	295031	**PRA:**	No		
Safety Function(s):	10	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

EXAMINATION ANSWER KEY

2016 RO NRC TEST

43 **ID: 1248407** **Points: 1.00**

Which of the following states the potential impact on RPV water level instrumentation from elevated Drywell temperatures?

	Affected RPV Water Level Instruments	Effect
A.	Yarways **ONLY**	May result in LOW indicated water level
B.	NR GEMACs **ONLY**	May result in HIGH indicated water level
C.	All	May result in LOW indicated water level
D.	All	May result in HIGH indicated water level

Answer: D

Answer Explanation			
K&A	295028 - High Drywell Temperature EK2.03 - Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Reactor water level indication (3.6/3.8)		
Level: RO	**Tier:** 1		**Group:** 1
References	EOP Users Guide	EMG-SP28	
Explanation	Proposed Answer: D Explanation: In accordance with the EOP Users Guide, all RPV water level instruments have a reference leg inside the drywell. When the drywell temperature is elevated, this results in heating of the reference legs and reducing the water density in the legs. As a result, RPV water level instruments will indicate a false high water level. A. Plausible if the applicant doesn't know that both sets of instruments have reference legs. B. Plausible if the applicant doesn't know that both sets of instruments have reference legs. C. Plausible if the applicant doesn't understand the impact of elevated reference leg temperature on indication.		
Lesson Plan Learning Objective/	2621.845.0.02 - PRIMARY CONTAINMENT CONTROL LP PCC-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status		
References Provided	**ILT:** None		**LORT:** Open
Question Source (New, Modified, Bank)	Bank		
Previous 2 NRC Exams (ILT Only)	No		

EXAMINATION ANSWER KEY

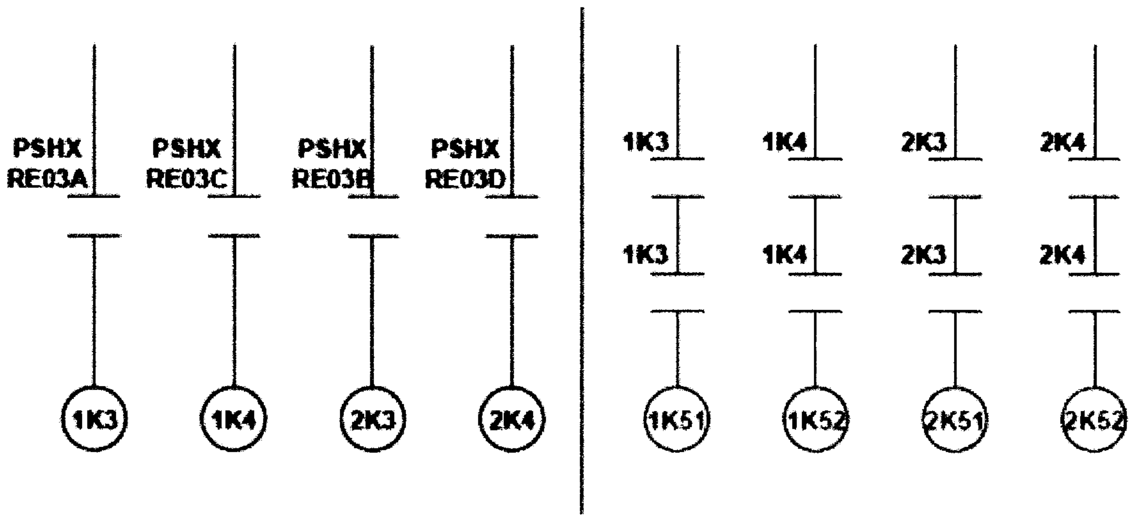
2016 RO NRC TEST

Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295028	**PRA:**	No	
Safety Function(s):	10	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

44 **ID: 1248408** **Points: 1.00**

The RPS scram logic for RPV high pressure is provided below.

With the reactor at rated power, which of the following will result in a full reactor scram from high RPV pressure?



A. Relays 1K3 **AND** 2K3 are **ENERGIZED**.

B. Relays 1K51 **AND** 1K52 are **DE-ENERGIZED**.

C. Contact PSHX RE03B **OR** contact PSHX RE03D **OPEN**.

D. Contact PSHX RE03C **AND** contact PSHX RE03B **OPEN**.

Answer: D

Answer Explanation			
K&A	295025 - High Reactor Pressure EK2.01 - Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following: RPS (4.1/4.1)		
Level: RO	**Tier:** 1		**Group:** 1
References	GE 237E566 sh. 1, 3, 5, 6		

Explanation	Proposed Answer: D Explanation: With the plant at power, all PSHX contacts are closed and ALL shown relays are energized. Relays 1K51 and 1K52 are RPS1 scram relays, and 2K51 and 2K52 are RPS2 scram relays. A full scram requires one of the RPS1 relays AND one of the RPS2 relays to be de-energized. With contacts PSHX RE03C and PSHX RE03B open, this will result in de-energizing relays 1K52 (RPS1) and 2K51 (RPS2) which results in a full scram. A. Plausible if the applicant believes RPS scram logic is energize to function. Some aspects of RPS are energize to function, such as ATWS circuitry. B. Plausible if the applicant believes RPS scram logic is 2 out of 2 taken once logic. Some aspects of RPS are 2 out of 2 taken once logic, such as ATWS logic. C. Plausible – This combination would require both contacts to be open. One or the other would not be sufficient to cause a full reactor scram.			
Lesson Plan **Learning Objective/**	2621.828.0.0037 - Reactor Protection System RPS-10441 - Given the system logic/electrical drawings, describe the system trip signals, setpoints and expected system response including power loss or failed components.			
References Provided	**ILT:** None			**LORT:** Open
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295025	**PRA:**	No	
Safety Function(s):	10	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

45 **ID: 1248409** **Points: 1.00**

The plant is at rated power. The following conditions exist:

- 1-1 Air Compressor is the LEAD Compressor
- 1-2 Air Compressor is tagged out of service
- 1-3 Air Compressor is the LAG Compressor

Plant events occurred at the following timeline:

At T=0 minutes: Annunciator FDR TO 460V 1A1 TRIP is received
At T=7 minutes: Due to an air leak, INST AIR SUPPLY PRESS indicates 73 psig and slowly lowering

What is the plant response regarding the Instrument and Service Air System components listed below at T=7 minutes?

	1-3 Air Compressor	V-6S-2, Service Air Isolation Valve
A.	Has Auto Started	Closes
B.	Has Auto Started	Remains Open
C.	Manual Start Required	Closes
D.	Manual Start Required	Remains Open

Answer: A

Answer Explanation

K&A	295019 - Partial or Complete Loss of Instrument Air AK3.03 - Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Service air isolations: Plant-Specific (3.2/3.2)		
Level: RO	**Tier:** 1		**Group:** 1
References	RAP-M2b	ABN-35	334
Explanation	Proposed Answer: A Explanation: IAW RAP M-2-b, SVC AIR DISCH VLV CLOSED, Service Air isolation valve V-6S-2 closes automatically when Service Air system pressure drops to less than 75 psig (with Service Air Isolation switch in NORMAL). Additionally ABN-35 states when INST AIR SUPPLY PRESS is < 75#, then Confirm V-6S-2 is Closed. IAW 334, the 1-3 compressor will auto start if it's the LAG compressor when in normal after stop (Not in PTL) and receiver air pressure drops to 90 psig. B. Plausible if the applicant doesn't know the service air isolation valve automatically closes or the setpoint. C. Plausible if the applicant doesn't know the LAG compressor auto start setpoint. D. Plausible if the applicant doesn't know the setpoints for either condition.		
Lesson Plan Learning Objective/	2621.828.0.0043 - SERVICE, INSTRUMENT AND BREATHING AIR CAS-10444 - Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

References Provided	**ILT:** None			**LORT:** Open	
Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X	
10CFR55 Content	**55.41b**	5	**55.43b**		
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of lead changes, and operating limitations and reasons for these operating characteristics				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	295019	**PRA:**	No		
Safety Function(s):	11	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

46 **ID: 1248410** **Points: 1.00**

The plant is shutdown for a refuel outage with fuel moves in progress on the refuel floor.

The refuel floor SRO has just notified the Control Room that a fuel bundle has dropped onto the top of the reactor core. The Control Room Operator reports the following radiation monitor readings:

- Radiation Monitor B9 indicates 75 mr/hr
- Radiation Monitor C10 indicates 80 mr/hr
- Reactor Building Ventilation Exhaust Radiation Monitor 1 indicates 20 mr/hr

Based on the above conditions, which of the following states the status of the RB Ventilation System **AND** the reason for this system status?

	RB Ventilation System Status	Reason
A.	Trips and isolates **BUT** is manually restarted	To reduce refuel floor radiation levels as quickly as possible
B.	Trips and isolates **BUT** is manually restarted	To ensure the greatest amount of air dilution prior to discharge
C.	Trips and isolates **AND** remains isolated	The system is not designed for high temperature air
D.	Trips and isolates **AND** remains isolated	To ensure air is discharged through a filtration system

Answer: D

Answer Explanation			
K&A	295023 - Refueling Accidents AK3.03 - Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS : Ventilation isolation (3.3/3.6)		
Level: RO	**Tier:** 1		**Group:** 1
References	RAP-10F1f	USAR 6.5.1.1	RAP-10F2m / RAP-10F4m

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Explanation	Proposed Answer: D Explanation: The question describes a refuel accident during refueling. The indications provide the following information: radiation monitor B9 is above its setpoint (50 mr/hr) and starts a 2-minute delay until the normal RB vent system isolates and SGTS starts; the RB vent radiation monitor is above its setpoint (9 mr/hr) to immediately isolate the normal RB vent system and start SGTS. Therefore, the normal RB vent system is isolated and SGTS has started to ensure the radioactive atmosphere is discharged through a filtration system. A. Plausible – IAW the station procedures, if ONLY the refuel area radiation monitors B9 or C9 have isolated the normal RB vent system and SGTS initiated, then the EOP directs placing the normal RB vent system back in service. This makes distractors A and B plausible, but not correct and the correct answer less obvious. There is no procedural allowance to override the vent systems when the RB vent monitors cause a valid isolation. B. Plausible – IAW the station procedures, if ONLY the refuel area radiation monitors B9 or C9 have isolated the normal RB vent system and SGTS initiated, then the EOP directs placing the normal RB vent system back in service. This makes distractors A and B plausible, but not correct and the correct answer less obvious. There is no procedural allowance to override the vent systems when the RB vent monitors cause a valid isolation. C. Plausible – Because the radioactivity in the discharged air will be decaying, this decay results in a temperature increase and distractor C is plausible.			
Lesson Plan Learning Objective/	2621.828.0.0043 - SERVICE, INSTRUMENT AND BREATHING AIR CAS-10444 - Describe the interlock signals and setpoints for the affected system components and expected system response including power loss or failed components.			
References Provided	**ILT:** None			**LORT:** Open
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**	
10CFR55 Content	**55.41b**	5	**55.43b**	
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of lead changes, and operating limitations and reasons for these operating characteristics			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295023	**PRA:**	No	

Safety Function(s):	11	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

47 **ID: 1248411** **Points: 1.00**

The control room has been evacuated due to a fire. The fire has been extinguished. ABN-29, Plant Fires, requires the following ventilation systems shutdown prior to purging the control room.

- A and B 480V Switchgear Room Ventilation System
- A/B Battery Room, MG Set Room Ventilation System
- Chemistry Laboratory Ventilation System
- Reactor Building Ventilation System

According to ABN-29, the reason this action is taken is to prevent smoke and fumes purged from the control room from being brought into these areas, which could ______________.

A. prevent personnel access

B. cause damage to equipment

C. set off automatic fire suppression systems

D. cause a reaction with other hazardous materials

Answer: C

Answer Explanation			
K&A	600000 - Plant Fire On Site AK3.04 - Knowledge of the reasons for the following responses as they apply to PLANT FIRE ON SITE: Actions contained in the abnormal procedure for plant fire on site (2.8/3.4)		
Level: RO	**Tier:** 1		**Group:** 1
References	ABN-29		
Explanation	Proposed Answer: C Explanation: IAW ABN-29, when a Control Room fire is extinguished, shutting down the ventilation systems for 'A' and 'B' 480V Swgr Room, A/B Battery Room, MG Set Room, Chem Lab, and RB HVAC will prevent smoke and fumes purged from the Control Room from being brought into a Vital Area that contains an automatic fire suppression system (and water deluge). A. Plausible since this outcome could be prevented, however the question specifically asks the reason stated in ABN-29 B. Plausible since this outcome could be prevented, however the question specifically asks the reason stated in ABN-29 D. Plausible since this outcome could be prevented, however the question specifically asks the reason stated in ABN-29		
Lesson Plan **Learning Objective/**	2621.828.0.0019 - FIRE PROTECTION SYSTEM FPS-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, EOP and EOP support procedures, and EP procedures.		
References Provided	**ILT:** None		**LORT:** Open

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	10	**55.43b**		
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	600000	**PRA:**	No		
Safety Function(s):	11	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

48 **ID: 1248412** **Points: 1.00**

The plant was at rated power when an event occurred. Indications and investigations revealed the following:

- Battery Charger MG Set A Breaker has opened
- Battery A Main Breaker has opened

Which of the following states the proper function of a DC Distribution System Automatic Transfer Switch under the given conditions?

The power to 125 VDC Bus __(1)__ has automatically transferred to 125 VDC Bus__(2)__.

	(1)	(2)
A.	DC-F	DC-C
B.	DC-1	DC-C
C.	DC-2	DC-B
D.	DC-E	DC-B

Answer: D

Answer Explanation				
K&A	295004 - Partial or Complete Loss of D.C. Power AA1.01 - Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : D.C. electrical distribution systems (3.3/3.4)			
Level: RO	**Tier:** 1		**Group:** 1	
References	ABN-53	RAP-9XF4e		
Explanation	Proposed Answer: D Explanation: The question stem describes a loss of power to 125 VDC Bus DC-A (both the battery charger and battery become disconnected from the Bus). When this bus de-energizes, then automatic transfer switch DC-E swaps from DC-A as the source of input power to 125 VDC Bus DC-B. A. Plausible if the applicant doesn't remember specific DC power supplies. Bus DC-F normally receives power from Bus DC-C, which is not affected by the loss of DC-A. B. Plausible if the applicant doesn't remember specific DC power supplies. Bus DC-1 normally receives power from Bus DC-B, which is not affected by the loss of DC-A. C. Plausible if the applicant doesn't remember specific DC power supplies. Bus DC-2 normally receives power from Bus DC-C, which is not affected by the loss of DC-A.			
Lesson Plan Learning Objective/	2621.828.0.0012 - DC DISTRIBUTION DCD-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.			
References Provided	**ILT:** None		**LORT:** Open	

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295004	**PRA:**	No	
Safety Function(s):	11	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

49 **ID: 1248413** **Points: 1.00**

The plant was at rated power when an event resulted in a scram. The plant is currently cooling down with the Shutdown Cooling System (SDC). Current conditions are as follows:

- RPV water level is 181 inches above TAF and steady
- Recirculation Pump suction temperature is 265°F
- SDC Pump C is operating, with the other SDC Pumps unavailable
- Main Condenser vacuum indicates 8 in Hg

An electrical fault in the breaker cubicle for SDC C discharge valve V-17-57 causes the valve to close. RPV temperature starts to rise.

Under these conditions, which of the following methods (and reason for using that method) can be used to cooldown the RPV?

A. Isolation Condensers since using this method will preserve RPV water inventory.

B. The Turbine Bypass Valves since this is the preferred method for rejecting decay heat from the reactor.

C. Feed with CRD and Bleed with Reactor Water Cleanup System letdown since the hotwell can still be considered to be available.

D. Alternate shutdown cooling with Safety Valves and Core Spray since this is the method recommended by ABN-3, Loss of Shutdown Cooling.

Answer: C

<table>
<tr><th colspan="4">Answer Explanation</th></tr>
<tr><td>K&A</td><td colspan="3">295021 - Loss of Shutdown Cooling
AA1.04 - Ability to operate and/or monitor the following as they apply to LOSS OF SHUTDOWN COOLING : Alternate heat removal methods (3.7/3.7)</td></tr>
<tr><td>Level: RO</td><td colspan="2">Tier: 1</td><td>Group: 1</td></tr>
<tr><td>References</td><td>ABN-3</td><td>303</td><td></td></tr>
<tr><td>Explanation</td><td colspan="3">Proposed Answer: C

Explanation: The question stem describes a loss of main condenser vacuum followed by a total loss of Shutdown Cooling (SDC). ABN-3, Loss of SDC, describes several methods of alternate cooling. Feed (with CRD/Cond Pump) and Bleed (with RWCU letdown) are the only choices available due to the conditions in the question stem. The reason the RWCU letdown can be used is even with no condenser vacuum, the condenser is still considered intact and available. RWCU to the hotwell might reach 120-130F, however this is not steam conditions.

A. Plausible if the applicant does not recall that Isolation Condensers cannot be used when RPV water level is > 160 in TAF.
B. Plausible if the applicant does not recall that the Main Condenser is not capable of accepting steam with no vacuum since the Bypass Valves will be closed.
D. Plausible since this method is available if EMRVs were used instead of SRVs, which the distractor states. SRVs do not have the capability to be manually operated.</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Lesson Plan **Learning Objective/**	2621.828.0.0045 - SHUTDOWN COOLING SYSTEM SDC-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, EOP & EOP support procedures and EP Procedures.			
References Provided	**ILT:** None			**LORT:** Open
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295021	**PRA:**	No	
Safety Function(s):	11	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

50 **ID: 1248414** **Points: 1.00**

The plant was at 83% power, with the following conditions:

- Recirculation Pump E is OFF, with its control switch in PTL

An event occurred which required the Operator to complete a rapid power reduction to 65% power with recirculation flow.

When power was stable, the following annunciators alarmed:

- MN BRKR 1A TRIP
- MN BRKR 1A LKOUT TRIP
- BUS 1A UV

Which **ONE** of the following actions are required?

A. Manually insert CRAM rods due to reduced core flow.

B. Manually scram the reactor due to reduced core flow.

C. Manually scram the reactor due to reduced feedwater flow.

D. Manually reduce recirculation flow due to reduced feedwater flow.

Answer: B

<table>
<tr><th colspan="5">Answer Explanation</th></tr>
<tr><td>K&A</td><td colspan="4">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)</td></tr>
<tr><td>Level: RO</td><td colspan="3">Tier: 1</td><td>Group: 1</td></tr>
<tr><td>References</td><td>ABN-2</td><td>ABN-1</td><td colspan="2">301.2</td></tr>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: B

Explanation: The question stem describes a loss of 4160 VAC Bus 1A. Recirculation pumps A, C, and E (presently OFF), and feedwater/condensate pumps A are powered from this Bus. When power is lost to Bus 1A, one feedwater pump (A), one condensate pump (A), and two recirculation pumps (A and C) are lost. ABN-2, Recirculation System Failures, requires a manual scram if <3 recirculation pumps are running OR if multiple recirculation pumps trip.

A. Plausible if the applicant does not recognize how many pumps trip correctly and believes only one recirc pump tripped leaving only 3 recirc pumps running therefore inserting the CRAM Array is directed in ABN-2.
C. Plausible if the applicant does not recognize that multiple recirc pumps trip but believes that multiple feed pumps tripped therefore scramming on reduced feedwater flow would be directed per ABN-17
D. Plausible if the applicant does not recognize recirc pumps tripped and does recall that a feedwater pump trip therefor the action to reduce recirc flow is required by ABN-17.</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Lesson Plan **Learning Objective/**	2621.828.0.0038 - REACTOR RECIRCULATION SYSTEM RRS-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, EOP & EOP support procedures and EP procedures.			
References Provided	**ILT:** None			**LORT:** Open
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	7	**55.43b**	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295006	**PRA:**	No	
Safety Function(s):	11	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

51 **ID: 1248415** **Points: 1.00**

The turbine is a rated speed and NOT connected to the grid.

Which of the following would cause the Turbine Stop Valves **AND** the Reheat and Intercept Valves to close?

A. RPV water level reaches 170”.

B. Turbine vibrations peak at 11 mils.

C. Condenser vacuum lowers to 23” Hg.

D. Turbine speed reached 1999 RPM during over-speed testing.

Answer: D

<table>
<tr><th colspan="6">Answer Explanation</th></tr>
<tr><td>K&A</td><td colspan="5">295005 - Main Turbine Trip
AA2.01 - Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : Turbine speed (2.6/2.7)</td></tr>
<tr><td>Level: RO</td><td colspan="3">Tier: 1</td><td colspan="2">Group: 1</td></tr>
<tr><td>References</td><td colspan="2">RAP H7E, Q3b, J1b</td><td>625.4.001</td><td colspan="2">ABN-10</td></tr>
<tr><td>Explanation</td><td colspan="5">Proposed Answer: D

Explanation: Closure of the turbine stop valves and reheat and intercept valves is indicative of a turbine trip. The over-speed trip is 1800 + 10% = 1980 RPM (+ 18 RPM in the overs-peed procedure).

A. Plausible – 170" is the setpoint for the alarm, not the turbine trip. An RPV water level of 175” will scram the reactor, which trips the turbine.
B. Plausible since there are procedural limitations on vibrations which can require a manual trip. However, there is no auto turbine trip from vibrations.
C. Plausible if the applicant does not know the setpoint. A condenser vacuum of 22” will both scram and trip the turbine.</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="5">2621.828.0.0051 - TURBINE CONTROLS
TCS-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td colspan="2">LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="5">Modified</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="5">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td>X</td><td>Comprehension or Analysis</td><td colspan="2"></td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>10</td><td>55.43b</td><td colspan="2"></td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295005	**PRA:**	No	
Safety Function(s):	11	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

52 **ID: 1248416** **Points: 1.00**

The reactor was at rated power when an event occurred. Current conditions include the following:

- RPV water level indicates 110" and rising slowly
- APRMs indicate 14% and lowering slowly
- Torus water temperature 105° F and rising slowly

The US has directed the performance of SP-22, Initiating the Liquid Poison System.

IAW the EOP User's Guide, initiation of Liquid Poison will achieve reactor shutdown prior to exceeding the...

A. Torus Load Limit

B. Heat Capacity Temperature Limit

C. Containment Spray Initiation Limit

D. Primary Containment Pressure Limit

Answer: B

Answer Explanation

K&A	295037 - SCRAM Condition Present and Reactor Power Above APRM Downscale or unknown EA2.07 - Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Containment conditions/isolations (4.0/4.2)		
Level: RO	**Tier:** 1		**Group:** 1
References	EOP Users Guide	RPV Control-With ATWS	
Explanation	Proposed Answer: B Explanation: The question stem describes a condition where the BIIT curve has been exceeded (14% power and 105F). The EOP User's Guide bases for initiating Liquid Poison when Torus Temperature cannot be maintained below the BIIT is to achieve a shutdown condition with the Hot Shutdown Boron Weight prior to exceeding the Heat Capacity Temperature Limit. A. Plausible – This is a valid plant parameter that is monitored during an ATWS and is a curve which is analyzed in the Primary Containment Control EOP. C. Plausible – This is a valid plant parameter that is monitored during an ATWS and is a curve which is analyzed in the Primary Containment Control EOP. D. Plausible – This is a valid plant parameter that is monitored during an ATWS and is a curve which is analyzed in the Primary Containment Control EOP.		
Lesson Plan Learning Objective/	2621.845.0.01B – RPV Control-With ATWS EWA-03055 – Given of copy of RPV Control, describe in detail each step or conditional statement, including technical basis, and how to perform each step as required.		
References Provided	**ILT:** None		**LORT:** Open

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	10	**55.43b**		
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	295037	**PRA:**	No		
Safety Function(s):	10	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

53 **ID: 1248417** **Points: 1.00**

The plant was at 22% power with generator output at 148 MWe, when an offsite electrical disturbance resulted in tripping the Main Transformer Lockout Relay (86T) and the following annunciator alarmed:

- GENERATOR - LKOUT RELAY TRIP

Which of the following states the impact on the reactor and the 230 KV Breakers GC1 and GD1?

	Reactor	230 KV Breakers GC1 and GD1
A.	Remains at power	Remain closed
B.	Automatically scrammed	Automatically tripped
C.	Required to be manually scrammed	Required to be manually tripped
D.	Remains at power	Automatically tripped

Answer: D

<table>
<tr><th colspan="5">Answer Explanation</th></tr>
<tr><td>K&A</td><td colspan="4">700000 - Generator Voltage and Electric Grid Disturbances
AA2.05 - Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Operational status of offsite circuit (3.2/3.8)</td></tr>
<tr><td>Level: RO</td><td colspan="2">Tier: 1</td><td colspan="2">Group: 1</td></tr>
<tr><td>References</td><td>RAP-R3d</td><td>ABN-10</td><td colspan="2">JC P6-50-00</td></tr>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: D

Explanation: The plant is at low power with the generator on-line, when an electrical disturbance resulted in an 86T relay trip. This relay then reasults in a main generator lockout and trip. When this occurs, breakers GC1 and GD1 will open. These are the main breakers that feed the offsite 230KV power circuit. The reactor remains at power (auto scram at >40%), and there no procedural requirements to scram.

A. Plausible – The reactor does remain at power. However, the 230KV offsite power breakers will automatically open on the 86T lockout.
B. Plausible – GD1 and GC1 are automatically tripped. From a higher power level, the reactor would have automatically scrammed. When less than 30% power, a scram is not required.
C. Plausible – If reactor power was >30% a scram would be required.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.828.0.0016 - ELECTRICAL DISTRIBUTION
ACD-10441 - Given the system logic/electrical drawings, describe the system trip signals, setpoints and expected system response including power loss or failed components.</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">Bank</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	5	**55.43b**	
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of lead changes, and operating limitations and reasons for these operating characteristics			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	700000	**PRA:**	No	
Safety Function(s):	11	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

54 **ID: 1248418** **Points: 1.00**

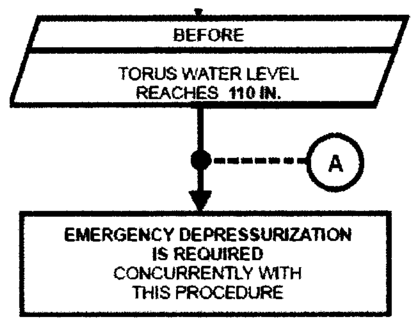
Which one of the following describes the requirement for an RPV Emergency Depressurization due to low Torus water level and the associated reason, in accordance with the Emergency Operating Procedures?

The Primary Containment Control EOP, states, "BEFORE Torus Water Level reaches ___(1)___ inches, Emergency Depressurization is required." This corresponds to the height of the ___(2)___.

	(1)	(2)
A.	110	Highest EMRV discharge line components
B.	110	Drywell vent header downcomer openings
C.	90	Highest EMRV discharge line components
D.	90	Drywell vent header downcomer openings

Answer: B

Answer Explanation			
K&A	295030 - Low Suppression Pool Water Level 2.4.6 - Knowledge of EOP mitigation strategies. (3.7/4.7)		
Level: RO	**Tier:** 1		**Group:** 1
References	EOP Users Guide		

Explanation	Proposed Answer: B Explanation: Primary Containment Control contains the following step:  Below 110 in., 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 airspace, a transient for which the Primary Containment is not designed. An Emergency RPV Depressurization is performed before 110 in. is reached, which transfers primary system energy to the Torus water to limit the consequences should a LOCA occur when Torus level drops below 110 in. A. Plausible – 110 inches is the correct level. However level for EMRV discharge device openings do not become a concern in the EOPs until 90 inches. C. Plausible –90 inches does corresponds to the EMRV discharge line components but per the EOPS you have to ED prior to 110 inches. D. Plausible – If the applicant believes 90 inches corresponds to the Drywell vent header downcomer openings therefore this would be plausible as it is basis for ED on low torus level but not at 90 inches.			
Lesson Plan Learning Objective/	2621.845.0.0056 - PRIMARY CONTAINMENT CONTROL LP PCC-03000 - Using Procedure EMG-3200.02, explain the basis for caution statements and evaluate plant conditions to determine that they are met			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**	
10CFR55 Content	**55.41b**	10	**55.43b**	
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			

EXAMINATION ANSWER KEY

2016 RO NRC TEST

System ID No.:	295030	PRA:	No	
Safety Function(s):	10	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

55 **ID: 1248419** **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 input into the starting circuit for the Core Spray System, have failed in their 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 Pump B auto starts as designed.

D. **NEITHER** Core Spray Pump A **NOR** B will auto start, but can be manually started. All other Core Spray components operate as designed.

Answer: A

Answer Explanation			
K&A	295024 - High Drywell Pressure 2.2.37 - Ability to determine operability and/or availablity of safety related equipment. (3.6/4.6)		
Level: RO	**Tier:** 1		**Group:** 1
References	RAP-C1f	NU 5060E6003, sh. 1-4	RAP-C2f
Explanation	Proposed Answer: A Explanation: PS-RV46A and PS-RV46B are safety related equipment and are linked to tech specs for the facility license that also feed into Core Spray starting logic. 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. A. Plausible – Two instrument failures in RPS could render an RPS channel inoperable, but the Core Spray start logic unique in that it is inter-mixed among systems. No manual actions are required for Core Spray to operate under the given conditions. C. Plausible – Two instrument failures in RPS could render an RPS channel inoperable, but the Core Spray start logic unique in that it is inter-mixed among systems. No manual actions are required for Core Spray to operate under the given conditions. D. Plausible – Two instrument failures in RPS could render an RPS channel inoperable, but the Core Spray start logic unique in that it is inter-mixed among systems. No manual actions are required for Core Spray to operate under the given conditions.		
Lesson Plan **Learning Objective/**	621.828.0.0010 - CORE SPRAY SYSTEM CSS-10439 - Given the system logic/electrical drawings, describe the system auto initiation signals, setpoints and expected system response including power loss or failed components.		

EXAMINATION ANSWER KEY

References Provided	**ILT:** None			**LORT:** Open	
Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	295024	**PRA:**	No		
Safety Function(s):	10	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

56 **ID: 1248420** **Points: 1.00**

The Control Room has been evacuated due to a fire, with the following conditions:

- The Reactor is scrammed
- The Turbine is tripped
- Control has been transferred to the Remote Shutdown Panel
- The RPV Pressure indicators on the Remote Shutdown Panel are damaged
- All systems are operating as designed

Which one of following describes what systems are specified in ABN-30, Control Room Evacuation, for maintaining RPV Pressure **FROM** the Remote Shutdown Panel, and where alternate RPV Pressure indications are available?

	System for Pressure Control	Alternate Pressure Indications
A.	'A' Isolation Condenser	RB 23' elevation near CRD
B.	'A' Isolation Condenser	RB 95' elevation near SLC
C.	'B' Isolation Condenser	RB 23' elevation near CRD
D.	'B' Isolation Condenser	RB 95' elevation near SLC

Answer: C

Answer Explanation			
K&A	295016 - Control Room Abandonment 2.4.11 - Knowledge of abnormal condition procedures. (4.0/4.2)		
Level: RO	**Tier:** 1		**Group:** 1
References	ABN-30		
Explanation	Proposed Answer: C Explanation: ABN-30, a NOTE at step 4.3.8 specifies the use of the 'B' Isolation Condenser from the remote shutdown panel and 'A' IC is only available to be controlled locally. Attachment ABN 30-9 provides information on available remote indications. Alternate Reactor Pressure indication is provided on RB 23' near CRD equipment. A. Plausible if the applicant doesn't remember which IC can be controlled from the RSP. B. Plausible if the applicant doesn't remember which IC can be controlled from the RSP. D. Plausible – The 'B' IC can be controlled. RB 95' elevation is listed in attachment ABN-30-9 as an alternate indication available, but not for reactor pressure.		
Lesson Plan **Learning Objective/**	2621.828.0.0023 - ISOLATION CONDENSERS ICS-10456 - DESCRIBE the Isolation Condenser System design feature which provides for the following: System control outside the control room (including automatic actions bypassed), Removal of non-condensable gases.		
References Provided	**ILT:** None		**LORT:** Open

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Question Source (New, Modified, Bank)	New				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	10	**55.43b**		
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	295016	**PRA:**	No		
Safety Function(s):	11	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

57 **ID: 1248421** **Points: 1.00**

An event has occurred which caused entry into EMG-3200.12, Radioactivity Release Control. This procedure includes the following Conditional Statement:

IF the release is from the Turbine Building,
THEN operate available Turbine Building ventilation per Support Procedure 51

Which of the following states the basis for this Conditional Statement?

A. To reduce the amount of radioactivity released.

B. To ensure a greater dilution factor during release.

C. Prevent an unmonitored ground release from the Turbine Building.

D. To ensure radioactivity release is ONLY through the Turbine Building Stack.

Answer: C

<table>
<tr><th colspan="5">Answer Explanation</th></tr>
<tr><td>K&A</td><td colspan="4">295038 - High Off-Site Release Rate
EK2.03 - Knowledge of the interrelations between HIGH OFF-SITE RELEASE RATE and the following: Plant ventilation systems (3.6/3.8)</td></tr>
<tr><td>Level: RO</td><td colspan="2">Tier: 1</td><td colspan="2">Group: 1</td></tr>
<tr><td>References</td><td>EMG 3200.12</td><td>EOP Users guide</td><td colspan="2">ENG-SP51</td></tr>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: C

Explanation: The EOP for radioactivity release control is entered when an alert emergency classification from offsite release rate has been declared. From the EOP User's Guide: "This Conditional Statement directs the operator to maintain the Turbine Building Ventilation System in service to preserve Turbine Building accessibility, and ensure that any radioactivity is discharged through a monitored release point, either the Main Stack for an elevated release, or via the Turbine Building Stack, which is considered a ground level release. When required, Support Procedure - 51 provides the necessary directions for restarting the Turbine Building Ventilation System." Some of the TB vent systems started discharge to the main stack (elevated release; ie., Exhaust Fan EF 1-7) and some to the TB stack (ground release; ie., exhaust fan EF 1-1).

A. Plausible if the applicant believes that guidance is meant to reduce the amount of radioactivity released.
B. Plausible if the applicant believes that guidance is meant to dilute the air prior to release.
D. Plausible if the applicant believes SP-51 only starts the turbine building fans that exhaust to the Turbine building stack and not the main stack as well.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.845.0.12 - Radioactivity Release Control LP
RRC-02483 - Using procedure Radioactivity Release Control, evaluate the technical basis for each step and apply this evaluation to determine the correct course of action under emergency conditions.</td></tr>
<tr><td>References Provided</td><td>ILT: None</td><td></td><td colspan="2">LORT: Open</td></tr>
</table>

EXAMINATION ANSWER KEY

Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features				
Justification for LORT K&A <3.0 values < 3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	295038	**PRA:**	No		
Safety Function(s):	10	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

58 **ID: 1248422** **Points: 1.00**

The plant was at 80% power. Recirculation Pump A has just been shutdown and the following valves are closed:

- PUMP SUCTION
- DISCHARGE
- DISCH BYPASS

IAW procedure 202.1, Power Operation, which one of the following limits is reduced due to the new operating loop configuration?

A. MCPR, as required by the fuel vendor.

B. FLLLP, as required by the USAR safety analysis.

C. MAPLHGR, as required by Technical Specifications.

D. MLHGR, as required by the Core Operating Limits Report.

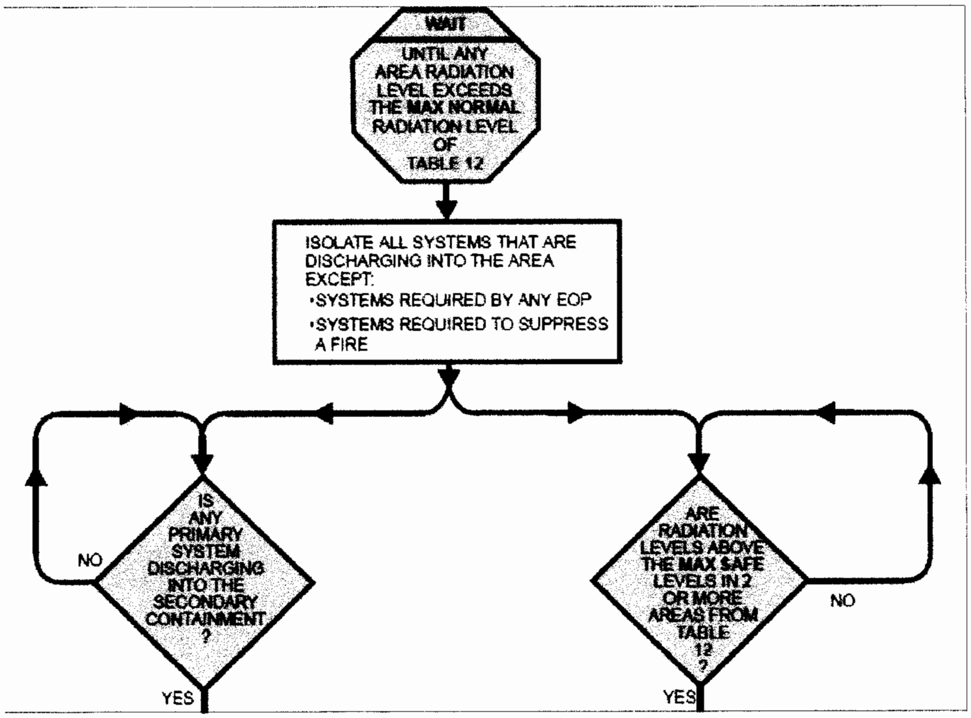
Answer: C

Answer Explanation			
K&A	295001 - Partial or Complete Loss of Forced Core Flow Circulation AK3.05 - Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Reduced loop operating requirements: Plant-Specific (3.2/3.6)		
Level: RO	**Tier:** 1		**Group:** 1
References	202.1	TS 3.3.F2.a.1	
Explanation	Proposed Answer: C Explanation: The question stem shows the plant at > 25% power and with one recirculation pump isolated. IAW Procedure 202.1, in this configuration, only MAPLHGR must be reduced from the normal 5-loop operating configuration to a 4-loop configuration, with power > 25% and the primary containment interted. A reduction in MAPLHGR is required by Technical Specifications 3.3.F.2.a.1. A. Plausible – This thermal limit does have penalties associated with it under certain conditions. However only MAPLHGR is the only choice affected by reduced core flow. B. Plausible – This thermal limit does have penalties associated with it under certain conditions. However only MAPLHGR is the only choice affected by reduced core flow. D. Plausible – This thermal limit does have penalties associated with it under certain conditions. However only MAPLHGR is the only choice affected by reduced core flow.		
Lesson Plan Learning Objective/	2621.828.0.0038 - REACTOR RECIRCULATION SYSTEM RRS-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.		
References Provided	**ILT:** None		**LORT:** Open

Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	5	**55.43b**		
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of lead changes, and operating limitations and reasons for these operating characteristics				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	295001	**PRA:**	No		
Safety Function(s):	11	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

59 **ID: 1248423** **Points: 1.00**

The plant was at rated power when entry into the Secondary Containment Control EOP was required. Note the Secondary Containment EOP section below.



IAW the EOP User's Guide, these steps are designed to terminate the increase in radiation levels above the MAX NORMAL values. The MAX NORMAL values are those radiation levels above which ____(1)____ and the MAX SAFE values are based on ____(2)____.

	(1)	(2)
A.	warn of a potential breach within the Secondary Containment.	Secondary Containment design limit
B.	could result in the failure of instrumentation necessary for safe shutdown of the plant	Secondary Containment design limit
C.	warn of a potential breach within the Secondary Containment.	Personnel Access
D.	could result in the failure of instrumentation necessary for safe shutdown of the plant	Personnel Access

Answer: C

Answer Explanation

<table>
<tr><td>K&A</td><td colspan="5">295033 - High Secondary Containment Area Radiation Levels
EK1.02 - Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Personnel protection (3.9/4.2)</td></tr>
<tr><td>Level: RO</td><td colspan="3">Tier: 1</td><td colspan="2">Group: 2</td></tr>
<tr><td>References</td><td>EOP User's Guide</td><td></td><td colspan="3"></td></tr>
<tr><td>Explanation</td><td colspan="5">Proposed Answer: C

Explanation: In accordance with the EOP User's Guide, Max Normal values provide warning of the onset of a potential breach or abnormal condition within the Secondary Containment. Max Safe values are defined to be the highest value in a specific area at which neither (1) equipment necessary for the safe shutdown of the plant will fail nor (2) personnel access necessary for the safe shutdown of the plant will be precluded.

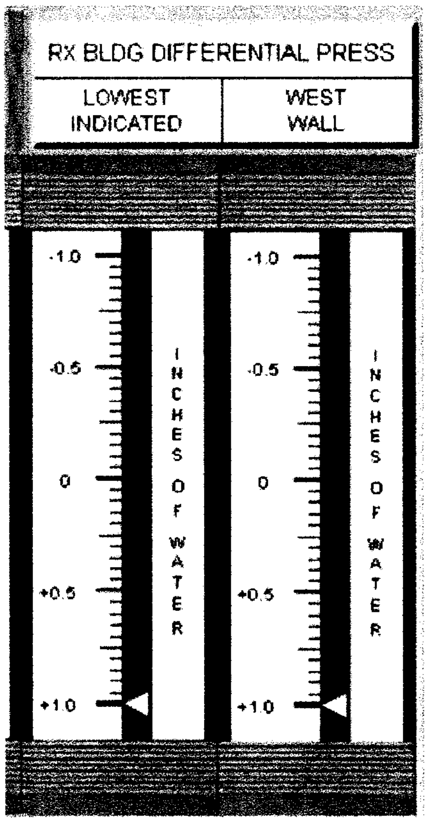
A. Plausible – The max normal value is designed to warn of a potential breach. However, the Max Safe values are based on personnel access or the operability of equipment required for safe shutdown. This is different that the secondary containment design limit.
B. Plausible – "result in the failure of instrumentation necessary for safe shutdown of the plant" is the basis for Max Safe Value, not Max Normal Value. Also, the Max Safe values are based on personnel access or the operability of equipment required for safe shutdown. This is different that the secondary containment design limit.
D. Plausible – A basis for Max Safe value is personnel access. "result in the failure of instrumentation necessary for safe shutdown of the plant" is also a basis for Max Safe Value, not Max Normal Value.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="5">2621.845.0.11SECONDARY CONTAINMENT CONTROL LP
SCC-03082 - Using Procedure 3200.11, evaluate the technical basis for each step and apply this evaluation to determine the correct course of action under emergency conditions.</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td colspan="2">LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="5">Modified</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="5">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td colspan="2">X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>10</td><td>55.43b</td><td colspan="2"></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="5">Administrative, normal, abnormal, and emergency operating procedures for the facility.</td></tr>
<tr><td>Justification for LORT K&A <3.0</td><td colspan="5">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="5">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="5">1</td></tr>
<tr><td>System ID No.:</td><td>295033</td><td>PRA:</td><td>No</td><td colspan="2"></td></tr>
</table>

Safety Function(s):	10	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

60 **ID: 1248424** **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

Answer: A

Answer Explanation	
K&A	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)

EXAMINATION ANSWER KEY

2016 RO NRC TEST

<table>
<tr><td>Level: RO</td><td colspan="4">Tier: 1</td><td colspan="3">Group: 2</td></tr>
<tr><td>References</td><td colspan="2">329</td><td colspan="2">330</td><td colspan="3"></td></tr>
<tr><td>Explanation</td><td colspan="7">Proposed Answer: A

Explanation: With RB Dp at +1.0 inches/water, 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.

B. Plausible if the applicant thinks the input to RB HVAC tripping is the same input to initiate SGTS.
C. Plausible - SGTS will remain in standby. The applicant needs to know the trip setpoint for RB HVAC.
D. Plausible - RB HVAC and SGTS can run simultaneously. Plausible if the applicant doesn't know the proper setpoints.</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="7">2621.828.0.0042 - SECONDARY CONTAINMENT AND SGTS
SGT-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.</td></tr>
<tr><td>References Provided</td><td colspan="3">ILT: None</td><td colspan="2"></td><td colspan="2">LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="7">Bank</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="7">No</td></tr>
<tr><td>Cognitive Level</td><td colspan="2">Memory or Fundamental Knowledge</td><td>X</td><td colspan="2">Comprehension or Analysis</td><td colspan="2"></td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td colspan="2">7</td><td colspan="2">55.43b</td><td colspan="2"></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="7">Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features</td></tr>
<tr><td>Justification for LORT K&A <3.0</td><td colspan="7">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="7">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="7">1</td></tr>
<tr><td>System ID No.:</td><td>295035</td><td colspan="2">PRA:</td><td colspan="2">No</td><td colspan="2"></td></tr>
<tr><td>Safety Function(s):</td><td>10</td><td colspan="2">☒ ILT</td><td colspan="2"></td><td colspan="2"></td></tr>
<tr><td>Category(s) (LORT Only):</td><td>N/A</td><td colspan="2">☐ LORT</td><td colspan="2"></td><td colspan="2"></td></tr>
</table>

61 **ID: 1248425** **Points: 1.00**

The plant was at rated power when condenser vacuum began to degrade uncontrollably.

Which one of the following describes the plant response as vacuum continues to degrade?

When condenser vacuum degrades to ____(1)____ inches, the ____(2)____ will close to prevent over-pressurizing the condenser

	(1)	(2)
A.	10	MSIVs
B.	10	Turbine Bypass Valves
C.	22	MSIVs
D.	22	Turbine Bypass Valves

Answer: B

Answer Explanation				
K&A	295002 - Loss of Main Condenser Vacuum AK3.04 - Knowledge of the reasons for the following responses as they apply to LOSS OF MAIN CONDENSER VACUUM : Bypass valve closure (3.4/3.6)			
Level: RO	**Tier:** 1		**Group:** 2	
References	RAP-Q1c	RAP-J1b		
Explanation	Proposed Answer: B Explanation: As condenser vacuum lowers, its ability to function as the ultimate heat sink also drops. At 22", a turbine trip and a scram signal are generated. At 10" vacuum, the turbine bypass valves are auto closed to prevent a main condenser over-pressure condition. When the condenser is over-pressurized, the condenser will relieve to the Turbine Building (atmospheric reliefs function at 5 psig). A. Plausible – The MSIVs will eventually go shut as condenser vacuum continues to degrade. However, at 10 inches, the TBVs go shut. C. Plausible – 22 inches is the turbine trip setpoint, where the turbine stop valves and turbine control valves go shut. The applicant needs to know the TBVs and MSIVs do not go shut on a turbine trip. D. Plausible – 22 inches is the turbine trip setpoint, where the turbine stop valves and turbine control valves go shut. The applicant needs to know the TBVs and MSIVs do not go shut on a turbine trip.			
Lesson Plan **Learning Objective/**	2621.828.0.0050TURBINE AND TURBINE AUXILIARIES MTA-10444 - Describe the interlock signals and setpoints for the affected system components (Main Turbine, Turbine Lube Oil) and expected system response including power loss or failed components.			
References Provided	**ILT:** None		**LORT:** Open	

Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**		
10CFR55 Content	**55.41b**	5	**55.43b**		
10CFR55 Explanation	Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of lead changes, and operating limitations and reasons for these operating characteristics				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	295002	**PRA:**	No		
Safety Function(s):	11	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

62 **ID: 1248426** **Points: 1.00**

The plant was at rated power when a leak occurred in the Drywell. The following plant data was obtained from SPDS.

- Reactor Pressure had reached a value as high as 1100 psig
- Drywell Pressure currently indicates 3.3 psig
- Reactor Power currently indicates 48%

Which of the following indications are correct for the current plant conditions?

1. Core Spray Main Pumps NZ01A **AND** NZ01B indicate **RED LIGHT ON**
2. MSIVs indicate **GREEN LIGHT ON**
3. Reactor Water Cleanup System isolation valves indicate **RED LIGHT ON**
4. DWEDT and DW Floor Sump isolation valves indicate **GREEN LIGHT ON**

A. 3 ONLY

B. 1 and 4

C. 1 and 2

D. 2, 3, and 4

Answer: B

Answer Explanation			
K&A	295010 - High Drywell Pressure AA1.02 - Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE : Drywell floor and equipment drain sumps (3.6/3.6)		
Level: RO	**Tier:** 1		**Group:** 2
References	EMG-SP1		
Explanation	Proposed Answer: B Explanation: RPV pressure had risen above the high pressure scram setpoint and reactor power remains at 48% on APRMs. The only way this amount of power can be produced, is if not all control rods are fully inserted. SPDS also showed that drywell pressure is 3.3 psig, which is above the high DW pressure scram and isolation setpoint, and Primary Containment Control EOP entry condition. This should result in an isolation of RWCU and DWEDT/DW Floor Sump. Also on a high DW pressure condition, core spray will automatically initiate (main/booster pumps A and B). Only answer B lists the correct indications. A. Plausible if the applicant doesn't recognize the RWCU isolation signal has not been reached. C. Plausible – Core Spray pumps will be running. The applicant needs to recognize there is no MSIV isolation signal is present. D. Plausible – DW equipment drain and floor drain sumps will isolate. The applicant needs to recognize there is no MSIV isolation signal is present and there is a RWCU isolation signal present.		
Lesson Plan Learning Objective/	2621.828.0.032 - PRIMARY CONTAINMENT PCS-00394 - Given auto isolation signals, list or identify causes(s), system response, and affected Primary Containment System components.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

References Provided	**ILT:** None			**LORT:** Open	
Question Source (New, Modified, Bank)	Modified				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X	
10CFR55 Content	**55.41b**	7	**55.43b**		
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	295010	**PRA:**	No		
Safety Function(s):	11	☒ **ILT**			
Category(s) (LORT Only):	N/A	☐ **LORT**			

EXAMINATION ANSWER KEY

2016 RO NRC TEST

63 **ID: 1248427** **Points: 1.00**

The plant is operating at 80% power when outboard MSIV NS-04B spuriously closes.

Which one of the following describes the automatic plant response to this transient over the next minute? (Assume no operator action)

A. Reactor power and pressure rise and stabilize at approximately 85% and 1030 psig, respectively.

B. Reactor power and pressure rise and stabilize at approximately 92% and 1045 psig, respectively.

C. The Reactor scrams and Reactor pressure is controlled by Turbine Bypass Valves.

D. The Reactor scrams and Reactor pressure is controlled by Isolation Condensers and/or EMRVs.

Answer: D

Answer Explanation			
K&A	295020 - Inadvertent Containment Isolation AA2.03 - Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION : Reactor power (3.7/3.7)		
Level: RO	**Tier:** 1		**Group:** 2
References	RAP-J2a	RAP-J3a	

<table>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: D

Explanation: Closure of MSIV NS-04B isolates one of the two main steam lines and causes a half scram. The remaining main steam line is rated for the steam flow equal to approximately 50% Reactor power. Since Reactor power was originally at 80%, the rise in steam flow through the other main steam line will cause a high flow condition (75%/50% > 120% isolation setpoint) and subsequent isolation of the second main steam line. This will cause a full Reactor scram.
Additionally, with both of the two main steam lines isolated, Turbine Bypass Valves will not be available to automatically control Reactor pressure (loss of normal heat sink). Isolation Condensers have a dedicated steam nozzle, therefore they are still available and will automatically initiate when Reactor pressure reaches approximately 1060 psig. EMRVs tap off the main steam lines inside of the MSIVs, therefore they are still available and will automatically initiate when/if Reactor pressure reaches 1085 psig.

A. Plausible – An automatic Reactor scram will occur based on MSIV position when the other main steam line isolates on high flow. This distractor is plausible if the candidate believes the remaining main steam line is capable of passing 80% of rated steam flow without isolating, as may be the case at a plant with four main steam lines.
B. Plausible – An automatic Reactor scram will occur based on MSIV position when the other main steam line isolates on high flow. This distractor is plausible if the candidate believes the remaining main steam line is capable of passing 80% of rated steam flow without isolating, as may be the case at a plant with four main steam lines.
C. Plausible – An automatic reactor scram will occur based on MSIV psotion when the other main steam line isolates on high flow. This distractor is plausible if the candidate believes the remains MSIV's remain open after the scram therefore pressure control will be controlled by the Turbine bypass valves.</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="4">2621.828.0.0026 - MAIN STEAM SYSTEM
MSS-10453 - Explain or describe how this system is interrelated with other plant systems.</td></tr>
<tr><td>References Provided</td><td colspan="2">ILT: None</td><td></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">New</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td>10</td><td>55.43b</td><td></td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Administrative, normal, abnormal, and emergency operating procedures for the facility.</td></tr>
<tr><td>Justification for LORT K&A <3.0</td><td colspan="4">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="4">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="4">1</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

System ID No.:	295020	**PRA:**	No	
Safety Function(s):	11	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

64 **ID: 1248428** **Points: 1.00**

The plant was shutdown with refueling activities in-progress. An event then occurred resulting in the following radiation-related annunciators (Panel 10F) alarming at time 0800 (hhmm):

- AREA MON - HI
- CRIT MON C5 HI
- NORTH WALL C10 - HI
- NORTH WALL C9 HI VENT TRIP
- OPER FLOOR B9 HI VENT TRIP
- VENT HI

At 0801, which of the following is correct?

A. The Standby Gas Treatment System is **NOT** yet in-service but shall be manually initiated IAW the Radioactivity Release Control EOP.

B. The Standby Gas treatment System has automatically initiated and shall remain in-service IAW the Secondary Containment Control EOP.

C. The normal Reactor Building Ventilation System has **NOT** yet isolated and shall remain in-service IAW the Secondary Containment Control EOP.

D. The normal Reactor Building Ventilation System has automatically isolated but shall be placed back in service IAW 329, Reactor Building Heating Cooling and Ventilation System.

Answer: B

Answer Explanation			
K&A	295034 - Secondary Containment Ventilation High Radiation 2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm. (4.1/4.3)		
Level: RO	**Tier:** 1	**Group:** 2	
References	Secondary Containment Control EOP	RAP-10F1f	

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Explanation	Proposed Answer: B Explanation: An event occurred during refueling activities and several refuel floor ARMs indicate above their high setpoint, and at least 1 of the 2 RB ventilation radiation monitors indicate above their high setpoint at 0800. With a single vent radiation monitor above its high setpoint, the Standby Gas Treatment System (SGTS) will immediately auto initiate and the normal RB Ventilation System will trip and isolate. If only the refuel floor ARMs were indicating above their high setpoint, the Secondary Containment Control EOP allows securing SGTS and reestablishing the normal ventilation system. But with the vent radiation monitors above their high setpoint, then IAW the Secondary Containment Control EOP, SGT shall remain in-service. A. Plausible – The applicant needs to understand auto initiation signals. Because the SGTS has already auto initiated, answer A becomes incorrect. C. Plausible – Some initiation signals will initiate SGTS without isolating RBV. In this case, RBV will isolate. The refuel floor ARMs can also auto initiate SGTS, however, that's after a 2-minute time delay. Because the normal RB Ventilation System did already isolate, then answer C is incorrect. D. Plausible – The normal RB Ventilation System did isolate, but it shall NOT be placed back into service.			
Lesson Plan **Learning Objective/**	2621.845.0.11SECONDARY CONTAINMENT CONTROL LP SCC-03082 - Using Procedure 3200.11, evaluate the technical basis for each step and apply this evaluation to determine the correct course of action under emergency conditions.			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	10	**55.43b**	
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295034	**PRA:**	No	
Safety Function(s):	10	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

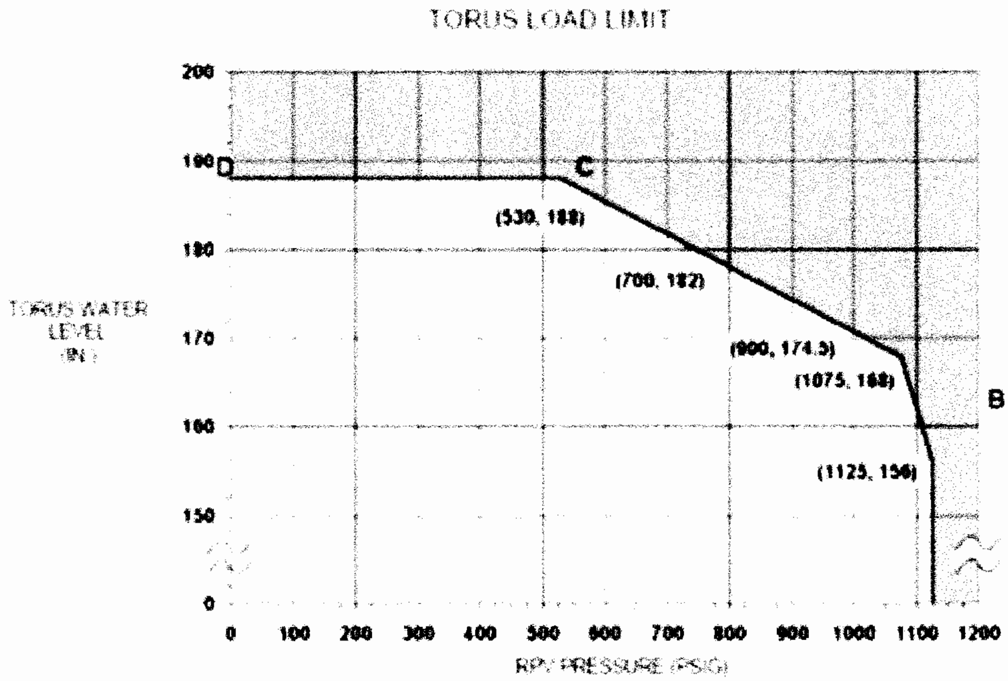
EXAMINATION ANSWER KEY

2016 RO NRC TEST

65 **ID: 1248429** **Points: 1.00**

The plant is operating at 100% power. An equipment malfunction resulted in Torus Water Level being raised to 178 inches.

Based on the Torus Load Limit Curve ONLY, which one of the following describes the potential plant impact of these conditions?



A. Torus structural support failure due to the weight of the torus water.

B. Primary Containment failure due to stresses at the saddle top flange to Torus shell weld.

C. An open EMRV can exceed the code allowable stresses and result in Primary Containment failure.

D. Primary Containment failure due to inability to vent decay heat.

Answer: C

Answer Explanation			
K&A	295029 - High Suppression Pool Water Level EK1.01 - Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Containment integrity (3.4/3.7)		
Level: RO	**Tier:** 1	**Group:** 2	
References	EOP Users guide		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Explanation	Proposed Answer: C Explanation: As supported in the reference, a high torus water level could result in the failure of EMRV components (tail pipe, pipe supports, quencher or quencher supports) during EMRV operation. Failure of the tail pipe could release steam directly to the DW, bypassing the torus suppression function and potentially failing the DW A. Plausible – Water loading is a concern, however, torus support structure is designed to withstand the water loading at 178 inches. B. Plausible – Basis for PCPL curve, not TLL curve – need to understand the difference in basis. D. Plausible – This is part of the basis for the MPCWLL curve, not the TLL curve.			
Lesson Plan Learning Objective/	2621.845.0.0056 - PRIMARY CONTAINMENT CONTROL PCC-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**	10	**55.43b**	
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295029	**PRA:**	No	
Safety Function(s):	10	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

66 **ID: 1248430** **Points: 1.00**

Which one of the following describes the correct sequence and flowpath to ensure the explosive limit is not reached for initially replacing air in the Main Generator with hydrogen during a plant startup?

A. H2 in through the upper header, air vented through the lower header.

B. H2 in through the lower header, air vented through the upper header.

C. CO2 in through the upper header, air vented through the lower header. H2 is then admitted through the lower header and the CO2 is vented through the upper header.

D. CO2 in through the lower header, air vented through the upper header. H2 is then admitted through the upper header and the CO2 is vented through the lower header.

Answer: D

Answer Explanation			
K&A	2.1.26 - Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen and hydrogen). (3.4/3.6)		
Level: RO	**Tier:** 3		**Group:**
References	336.3	SA-CE-116-1003	
Explanation	Proposed Answer: D Explanation: CO2 is admitted through the lower header (heavier than Air). The CO2 will fill the generator and push the Air out the upper header. This ensures that all air is removed from the generator prior to adding H2 to the generator to ensure the explosive limit is not reached. H2 is then admitted through the upper header (lighter than CO2), CO2 is vented through the lower header. A. Plausible if the applicant doesn't know the process for avoiding the explosive mixture of H2 and air. Must use CO2 as a buffer. B. Plausible if the applicant doesn't know the process for avoiding the explosive mixture of H2 and air. Must use CO2 as a buffer. C. Plausible – CO2 is admitted through the lower header (heavier than Air). The CO2 will fill the generator and push the Air out the upper header. H2 is then admitted through the upper header (lighter than CO2), CO2 is vented through the lower header		
Lesson Plan **Learning Objective/**	2621.828.0.0067 - GENERATOR AUXILIARIES GAX-10446 - Identify and explain system operating controls / indications (Seal Oil, Hydrogen Gas, Stator Water Cooling, Bus Duct Cooling) under all plant operating conditions		
References Provided	**ILT:** None		**LORT:** Open
Question Source (New, Modified, Bank)	New		
Previous 2 NRC Exams (ILT Only)	No		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Cognitive Level	**Memory or Fundamental Knowledge**	X	**Comprehension or Analysis**	
10CFR55 Content	**55.41b**	10	**55.43b**	
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	245000	**PRA:**	No	
Safety Function(s):	4	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

67 **ID: 1248431** **Points: 1.00**

Which of the following states the expectation for informing the US of alarm annunciation in accordance with OP-AA-103-102, Watch-Standing Practices?

The information below was covered during the brief:
- All expected alarms were covered
- All appropriate RAPS were reviewed

When an **alarm associated** with the evolution comes in ___1___, and a log entry __2__ required for the alarm.

	1	2
A.	the US does not need to be informed if the alarm was briefed and is flagged	is not
B.	the US must be informed if the alarm was **not** briefed and **not** flagged	is
C.	the US must be informed if the alarm was briefed and is flagged	is not
D.	the US does not need to be informed if the alarm was briefed and **not** flagged	is

Answer: A

Answer Explanation			
K&A	2.1.1 - Knowledge of conduct of operations requirements. (3.8/4.2)		
Level: RO	Tier: 3		Group:
References	OP-OC-101-111-1001		
Explanation	Proposed answer: A Explanation: IAW procedure OP-AA-103-102, watchstanding principles, when an expected alarm comes and it is briefed and flagged the US does not have to be informed. A log entry is also not required for expected alarms that were briefed B. Plausible since the US must be informed because the alarm was not briefed but there is no requirement for a log entry to be made because it was an alarm associated with the evolution. If the applicant believes that since unexpected alarms need to be logged this would be a correct answer since it was not briefed. C. Plausible if the applicant believes that since the alarm has been briefed and flagged then the US does not need to be informed because he is aware of the condition already and a log entry is to be made due to alarms are logged when they are received. But no log entry is required due to it being associated with the planned evolution. D. Plausible if the applicant believes that since the alarm has been briefed then the US does not need to be informed because he is aware of the conditions already. A log entry is not required due to it being associated with the planned evolution.		
Lesson Plan **Learning Objective/**	2621.dbig.0033 - Watchstanding Practices CT1 - Following this lesson, the student will be able to demonstrate understanding of the application of Watchstanding Practices in accordance with OP-AA-103-102.		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

References Provided	ILT: None			LORT: Open	
Question Source (New, Modified, Bank)	New				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	Memory or Fundamental Knowledge	X	Comprehension or Analysis		
10CFR55 Content	55.41b	10	55.43b		
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	N/A	PRA:	No		
Safety Function(s):	14	☒ ILT			
Category(s) (LORT Only):	N/A	☐ LORT			

68 **ID: 1248433** **Points: 1.00**

The plant is starting up after a refuel outage.

Which of the following states who can manipulate Reactor Controls?

A. An Equipment Operator candidate who is being directly supervised by an active licensed operator.

B. An active licensed operator with a corrective lenses license restriction who does not have his glasses.

C. An inactive licensed operator who is reactivating and who is being directly supervised by an active licensed operator.

D. A Reactor engineer who has been selected for the next initial license training class who is being directly supervised by an active licensed operator.

Answer: C

<table>
<tr><th colspan="4">Answer Explanation</th></tr>
<tr><td>K&A</td><td colspan="3">2.1.4 - Knowledge of individual licensed operator related to shift staffing, such as medical requirements, "no-solo" operation, maintenanc of active license status, 10CFR55, etc. (3.3/3.8)</td></tr>
<tr><td>Level: RO</td><td colspan="2">Tier: 3</td><td>Group:</td></tr>
<tr><td>References</td><td>OP-AA-103-103</td><td>OP-AA-105-102</td><td></td></tr>
<tr><td>Explanation</td><td colspan="3">Proposed Answer: C

Explanation: IAW the OP-AA-103-103, an inactive licensed operator must be enrolled in a license reactivation program to perform main control room manipulations. IAW OP-AA-105-102, the hours spent shift functions will be performed in the presence and under the direct supervision of an active RO or SRO. Therefore, the inactive operator must be reactivating and under the direct supervision of an active operator.

Note: This question matches the KA statement since the process for maintaining the configuration and status of reactor controls is related to who can change or manipulate the reactor controls. Procedures allow only certain individuals to manipulate the reactor controls (apparatus and mechanisms that the manipulation of would directly affect the reactivity or power level of the reactor).

A. Plausible – Trainees who are enrolled in a license training program can manipulate the controls while under direct supervision of a licensed operator. However, the candidate listed is not enrolled in a license training program.
B. Plausible – An active licensed operator would normally be a correct answer. However, it is the operators' responsibility to meet his license restrictions. The operator is only allowed to perform license functions when license restrictions are met.
D. Plausible – Trainees who are enrolled in a license training program can manipulate the controls while under direct supervision of a licensed operator. However, the candidate listed is not enrolled in a license training program</td></tr>
<tr><td>Lesson Plan Learning Objective/</td><td colspan="3">2621.830.0.0018 - Equipment Control – Admin
2.2.2 - Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 RO NRC TEST

References Provided	ILT: None			LORT: Open	
Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	Memory or Fundamental Knowledge	X	Comprehension or Analysis		
10CFR55 Content	55.41b	10	55.43b		
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	N/A	PRA:	No		
Safety Function(s):	14	☒ ILT			
Category(s) (LORT Only):	N/A	☐ LORT			

69 **ID: 1248434** **Points: 1.00**

The following conditions exist:

- Reactor coolant temperature is 180°F.
- All Reactor Manual Control interlocks are in normal condition
- Rod 02-27 is at position 12
- Rod 50-35 is being moved from position 00 to position 48 from panel 4F
- All other rods are at position 00

Which one of the following is the current Reactor Operating Condition, in accordance with Technical Specifications?

A. Refuel Mode

B. Startup Mode

C. Cold Shutdown Condition

D. Hot Shutdown Condition

Answer: B

Answer Explanation			
K&A	2.2.35 - Ability to determine Technical Specification Mode of Operation. (3.6/4.6)		
Level: RO	Tier: 3	Group:	
References	Technical Specifications - Definitions		
Explanation	Proposed Answer: B Explanation: The reactor is in the startup mode when the reactor mode switch is in the startup mode position and allows more than one rod to be withdrawn from the core to bring the reactor to power. Since the stem states that one rod is at position 12 which is allowed during refuel mode and a second rod is being moved out then the Rx mode must be in the startup mode with these conditions as the refuel mode will not allow a second rod to move when reactor manual control interlocks are in the normal condition. A. Plausible – As stated in the stem, with temprature at 180°F this temperature is around what it would be during the start of a startup and if the mod switch was in refuel it will allow only onerod to be withdrawn from the core all the way. It is plausible if the applicant believes that one rod can be moved to full out position while in the refuel mode while another rod is at position 12. C. Plausible since the reactor is not critical and is less than 212 degrees. Cold shutdown requires all rods inserted and less than 212 degrees. If the applicant just believes that since temperature is less than 212 and only one rod is withdrawn the reactor will stay sub-critical therefore the reactor is in the Cold Shutdown condition D. Plausible – This condition would be correct if temperature was greater than 212 degress and mode switch in shutdown. If the applicant believes that since one rod is being moved to full out then the reactor cannot be in a cold shutdown condition and therefore believes that it is in a hot shutdown because of the one rod being withdrawn.		

Lesson Plan **Learning Objective/**	2621.850.0.0090 - Overview/Highlights of Technical Specifications TSX-01920 - Given various plant indications (and their values) or copies of Control Room/Plant logs, evaluate the indications to determine plant status with respect to Operating License and Technical Specifications.			
References Provided	ILT: None		LORT: Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b	7	55.43b	
10CFR55 Explanation	Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features			
JJustification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	14	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

70 **ID: 1248435** **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 All 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			
K&A	2.3.14 - Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (3.4/3.8)		
Level: RO	Tier: 3		Group:
References	EMG-SP21		
Explanation	Proposed Answer: B Explanation: 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. A. Plausible – This is an alternate method to insert control rods during an ATWS. However, it will not raise RB airborne contamination levels. C. Plausible – This is an alternate method to insert control rods during an ATWS and makes RPS de-energize like a normal scram would. However, it will not raise RB airborne contamination levels. If the applicant does not understand the RPS prints and where the 100 amp breaker is located in the prints and believes that it will only insert the rods but not isolate the SDV from the RBEDT then is would be correct. D. Plausible – This is an alternate method to insert control rods during an ATWS and inserts rods and sends a signal to isolate the SDV . However, it will not raise RB airborne contamination levels. If the applicant does not understand the RPS prints and where the Subchannel test switches are located the prints and believes that it will only insert the rods but not isolate the SDV from the RBEDT then is would be correct.		
Lesson Plan Learning Objective/	2621.845.0.01B - RPV CONTROL-WITH ATWS EWA-03055 - Given a copy of RPV Control, describe in detail each step or conditional statement, including technical basis, and how to perform each step as required.		
References Provided	ILT: None		LORT: Open
Question Source (New, Modified, Bank)	Bank		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b	12	55.43b	
10CFR55 Explanation	Radiological safety principles and procedures			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	15	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

71 **ID: 1248436** **Points: 1.00**

The plant is at rated power. An EO is required to manipulate a manual valve (located at floor level, and requires no tools to manipulate) in a NON Self-Locking Locked High Radiation Area (LHRA). This area has a peak dose rate of 1050 mr/hr, and is routinely surveyed by Radiation Protection.

Which of the following steps are **REQUIRED** by the Operator IAW RP-AA-460, Controls for High and Locked High Radiation Areas after signing onto a RWP authorizing access to the LHRA ?

1. Receive a briefing from the RP Tech prior to entry
2. Ensure that the RP Tech accompanies you into the LHRA
3. Verify the maximum dose rate with your electronic dosimetry
4. Upon completion of work and prior to leaving the area, ensure access has been verified locked by an RP Tech or Access Control Guard

A. 1 and 2

B. 1 and 4

C. 3 and 4

D. 2, 3, and 4

Answer: B

Answer Explanation				
K&A	2.3.13 - Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. (3.8)			
Level: RO	Tier: 3			Group:
References	RP-AA-460			
Explanation	Proposed Answer: B Explanation: Of the 4 requirement choices listed, the only 2 required by RP-AA-460 are: 1. Receive a briefing from the RP Tech prior to entry; and 4. Upon completion of work and prior to leaving the area, ensure access has been verified locked by an RP Tech and Access Control Guard. A. Plausible – Choice 1 is correct. However, the applicant needs to know it is not required that the RP Tech accompanies you into the LHRA. C. Plausible – Choice 4 is correct. However, the applicant needs to know it is not required to verify the maximum dose rate with your electronic dosimetry. D. Plausible – Choice 4 is correct. However, the applicant needs to know it is not required that the RP Tech accompanies you into the LHRA or to verify the maximum dose rate with your electronic dosimetry.			
Lesson Plan **Learning Objective/**	2621.830.0.0015 - Radiation Control – Admin B2.3.12 - Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.			
References Provided	none			LORT: Open

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge	X	Comprehension or Analysis	
10CFR55 Content	55.41b	12	55.43b	
10CFR55 Explanation	Radiological Safety principles and procedures			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	9	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

72 **ID: 1248437** **Points: 1.00**

An electrical fire started inside the 'C' 4160V Switchgear Vault.

In accordance with ABN-29, Plant Fires, which of the following states the fire suppression agent and initiation method to suppress this fire?

	Suppression Agent	Initiation Method
A.	Halon 1301	Manual
B.	Dry pipe sprinkler	Confirm Automatic
C.	Portable CO2 Fire Extinguisher	Manual
D.	High pressure CO2	Confirm Automatic

Answer: C

Answer Explanation			
K&A	2.4.26 - Knowledge of facility protection requirements, including fire brigade and portable fire fighting equipment usage. (3.1/3.6)		
Level: RO	Tier: 3		Group:
References	ABN-29		
Explanation	Proposed Answer: C Explanation: ABN-29, section 4.4, directs the use of portable fire extinguishers for the 4160V 'C' switchgear vault. The low pressure CO2 system also protects the 4160 Volt 'C' switchgear vault and is manually initiated. ABN 29 directs low pressure CO2 be manually initiated if the portable CO2 extinguishers cannot be used to extinguish the fire. A. Plausible – This is an extinguishing agent for electric plant components. Halon protects 480 volt switchgear rooms A and B. B. Plausible – This is an extinguishing agent for electric plant components. Drypipe system protects the 4160 A and B vaults. D. Plausible – This is an extinguishing agent for electric plant components. Low pressure CO2 protects the 'C' switchgear vault. However it must be manually initiated. KA Match Justification - **Knowledge of facility protection requirements, including portable fire fighting equipment usage.** This question is testing the applicants' knowledge of plant fire procedures and when portable fire fighting equipment is required to be used.		
Lesson Plan **Learning Objective/**	2621.828.0.0019 - FIRE PROTECTION SYSTEM 286-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, EOP and EOP support procedures, and EP procedures.		
References Provided	ILT: None		LORT: Open

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Question Source (New, Modified, Bank)	Bank				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	Memory or Fundamental Knowledge	X	Comprehension or Analysis		
10CFR55 Content	55.41b	10	55.43b		
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.				
Justification for LORT K&A <3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	N/A	PRA:	No		
Safety Function(s):	11	☒ ILT			
Category(s) (LORT Only):	N/A	☐ LORT			

73 **ID: 1248438** **Points: 1.00**

The plant was at rated power when a turbine trip/reactor scram occurred. Plant conditions include the following:

- Both Isolation Condensers have auto initiated.
- Two (2) EMRV's have cycled OPEN.
- Isolation Condenser B level is 7.7 feet and rising
- Annunciator SHELL TEMP HI is in alarm
- Attempts to isolate the affected isolation condenser have failed
- Torus bulk temperature is 91 degrees F and steady
- NO other annunciators are in alarm

IN ADDITION TO RPV CONTROL - NO ATWS EOP, which EOP(s), if any, has (have) met entry conditions and require implementation?

A. None

B. Primary Containment Control EOP ONLY

C. Radioactivity Release Control EOP ONLY

D. Primary Containment Control EOP AND Radioactivity Release Control EOP

Answer: C

Answer Explanation			
K&A	2.4.4 - Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. (4.5/.7)		
Level: RO	Tier: 3		Group:
References	RR EOP	EOP Users Guide	
Explanation	Proposed Answer: C Explanation: The question stem provides indications of an Isolation Condenser Tube Leak. IAW the Radioactivity Release EOP, a confirmed IC tube leak requires entry into the RR EOP. A. Plausible if the applicant does not recognize the requirement to enter EOP RR for an IC tube leak. B. Plausible if the applicant does not know the EOP entry condition setpoint for Torus Temperature. Torus temperature is close to an entry setpoint. D. Plausible if the applicant does not know the EOP entry condition setpoint for Torus Temperature. Torus temperature is close to an entry setpoint.		
Lesson Plan Learning Objective/	2621.845.0.0058 - RRC-01667		
References Provided	ILT: None		LORT: Open
Question Source (New, Modified, Bank)	Bank		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b	10	55.43b	
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	10	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

74 **ID: 1248439** **Points: 1.00**

The plant is operating at 100% power when the VENT HI annunciator, RAP-10F1f, alarms.

Which one of the following describes the possible cause for this alarm?

A. RWCU leak in the Drywell

B. Recirculation Pump seal failure

C. Isolation Condenser tube leak

D. RWCU leak in the Reactor Building

Answer: D

Answer Explanation			
K&A	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	RAP-10F1F		
Explanation	Proposed Answer: D Explanation: The applicant must understand that the Vent Hi alarm is monitored in the Reactor Building Ventilation duct. A leak from RWCU inside the reactor building could cause ventilation radiation monitors to rise which causes the given annunciator. A. Plausible – This leak is contained in the drywell, and would only cause containment rad level changes. If the applicant believes that the Vent Hi alarm is sensed by the monitors in the Drywell then this would be a correct answer. B. Plausible – This leak is contained in the drywell, and would only cause containment rad level changes. If the applicant believes that the Vent Hi alarm is sensed by the monitors in the Drywell then this would be a correct answer. C. Plausible – If the applicant believes that since this leak is in the Reactor building then it could bring in the Vent Hi alarm as there are alarms for area radiation levels located at the IC's. However even though this leak is inside the Reactor building this leak would result in an IC rad alarm, it is not vented to the stack therefore the ventilation monitors in the Reactor building would not rise and the Vent Hi alarm would not come in.		
Lesson Plan Learning Objective/	2621.828.0.0042 – Secondary Containment and SGTS SGT-10449 - State the function and interpretation of system alarms, alone and in combination, as applicable in accordance with the system RAPS.		
References Provided	ILT: None		LORT: Open
Question Source (New, Modified, Bank)	New		
Previous 2 NRC Exams (ILT Only)	No		

EXAMINATION ANSWER KEY

2016 RO NRC TEST

Cognitive Level	Memory or Fundamental Knowledge	X	Comprehension or Analysis	
10CFR55 Content	55.41b	11	55.43b	
10CFR55 Explanation	Purpose and operation of radiation monitoring systems, including alarms and survey equipment			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	15	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

75 **ID: 1250433** **Points: 1.00**

Before performing a Technical Specification surveillance in the Reactor Building on the Core Spray system, whose permission is required to initiate a surveillance on this system?

A. Unit Supervisor

B. Shift Operations Superintendent

C. Shift Manager

D. Field Supervisor

Answer: A

Answer Explanation				
K&A	2.2.12 - Knowledge of surveillance procedures. (3.7/4.1)			
Level: RO	Tier: 3		Group:	
References	WC-AA-111	610. procedures		
Explanation	Proposed Answer: A Explanation: per WC-AA-111 the Surveillance test cover sheet requires a shift management signature to commence work. Per Core spray surveillances the Unit Supervisor must sign to commence work for these systems. This is pertinent to Reactor Operators to ensure they can verify they have the correct level of permission to perform the surveillance. B. Plausible if the applicant believes that since the Operations Director has overall responsibility for the surveillances assigned to the department and the work schedule is reviewed by the Ops Director/SOS. They are required to review and approve the work schedule but not required for initiation of a core spray surveillance. C. Plausible if the applicant believes that since the Shift Manager has responsibility of shift activities it must be the shift manager to grant permission to initiate a Tech Spec surveillance. D. Plausible if the applicant believes that since the Field Supervisor will be directly supervising surveillances conducted in the Reactor Building that they are the ones to grant permission to initiate work.			
Lesson Plan Learning Objective/	N-OC-2621.828.0.0010 –Core Spray System CSS-10447 - Given normal operating procedures and documents for the system, describe or interpret the procedural steps.			
References Provided	ILT: None		LORT: Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			

EXAMINATION ANSWER KEY

2016 RO NRC TEST

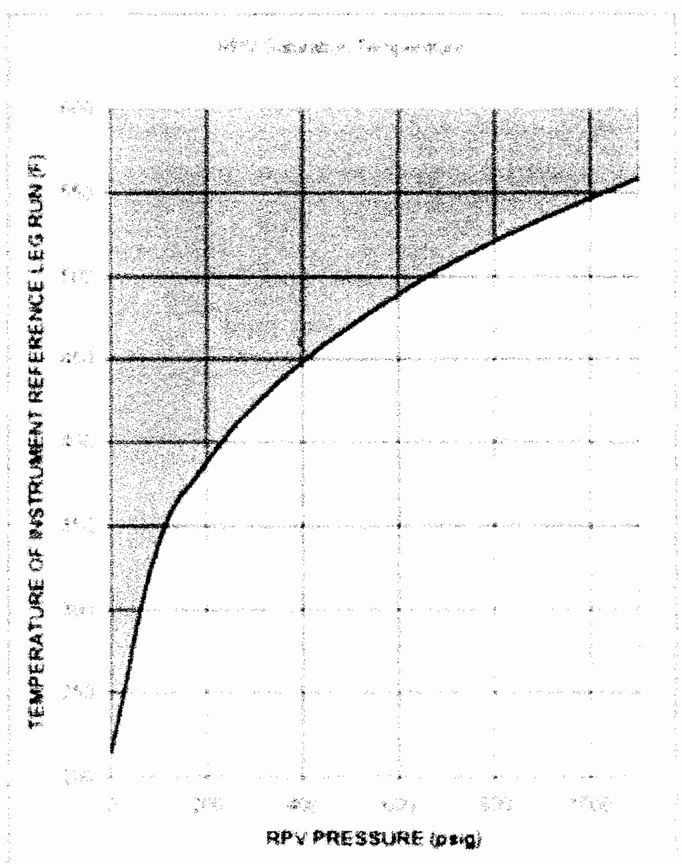
Cognitive Level	Memory or Fundamental Knowledge	X	Comprehension or Analysis	
10CFR55 Content	55.41b	10	55.43b	
10CFR55 Explanation	Administrative, normal, abnormal, and emergency operating procedures for the facility.			
Justification for LORT K&A <3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	3	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

76 **ID: 1248441** **Points: 1.00**

The plant was at rated power when an event occurred. Present plant conditions are as follows:

- ALL control rods indicate green background **EXCEPT** 4 control rods which indicate 04
- ALL RPV water level instrument reference leg temperatures indicate > 450°F
- RPV water level indicators GEMAC A, B, & C indicate downscale
- RPV pressure indicates 300 psig and lowering slowly
- REACTOR LEVEL FUEL ZONE indicators are **NOT** reliable
- Torus water level indicates 160" and steady

Which of the following actions is required?



A. Manually open all EMRVS IAW the RPV Flooding - No ATWS EOP.

B. Terminate and prevent RPV injection, **THEN** manually open all EMRVs IAW the RPV Flooding - With ATWS EOP.

C. Restore and maintain RPV water level between 100" and 175" using the Core Spray System IAW SP-4, Operation of the Core Spray System.

D. Restore and maintain RPV water level between 138" and 175" using Feedwater/Condensate IAW SP-2, Feedwater and Condensate System Operation.

Answer: A

Answer Explanation			
K&A	295031 - Reactor Low Water Level EA2.03 - Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : Reactor pressure (4.2)		
Level: SRO	**Tier:** 1		**Group:** 1
General References	EMG-SP28	RPV Flooding - No ATWS EOP	
Explanation	Proposed Answer: A Explanation: Indications show that the temperature in the Primary Containment is high, that RPV water level instruments Fuel Zone are unreliable, and all 3 NR GEMAC instruments indicate downscale. IAW SP28, all YARWAY and GEMAC RPV water level instrument reference leg temperatures place the instruments in the saturated Unsafe Region of the RPV Saturation Temperature Curve and cannot be used to determine RPV water level. With control rods at position 04, the reactor can still be determined to be shutdown. With the reactor shutdown, and no available RPV water level instruments, entry into the RPV Flooding - No ATWS is required and the SRO will direct that all EMRVs be opened. __Note:__ This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on testing the ability to assess a plant condition (shutdown under all conditions), to prescribe the correct procedure section (EMG-SP28 and RPV Flooding - No ATWS EOP). B. Plausible – If the applicant thinks that an ATWS is in progress, then answer B would be correct. The plant is not in an ATWS, therefore B is incorrect. C. Plausible – With the 3 NR GEMACs downscale, if the candidate does not realize the effect of the reference leg temperatures, they could conclude that restoring/maintaining RPV water level 138"-175" is correct. Incorrect since the level indicators are invalid. D. Plausible – With the 3 NR GEMACs downscale, if the candidate does not realize the effect of the reference leg temperatures, they could conclude that restoring/maintaining RPV water level 138"-175" is correct. Incorrect since the level indicators are invalid.		
Lesson Plan **Learning Objective/**	N-OC-2621.845.0.01A - RPV Control No ATWS ENA-10045 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.		
References Provided	**ILT:** None		**LORT:** Open

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**		**55.43b**	5
10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			
Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295031	**PRA:**	No	
Safety Function(s):	10	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

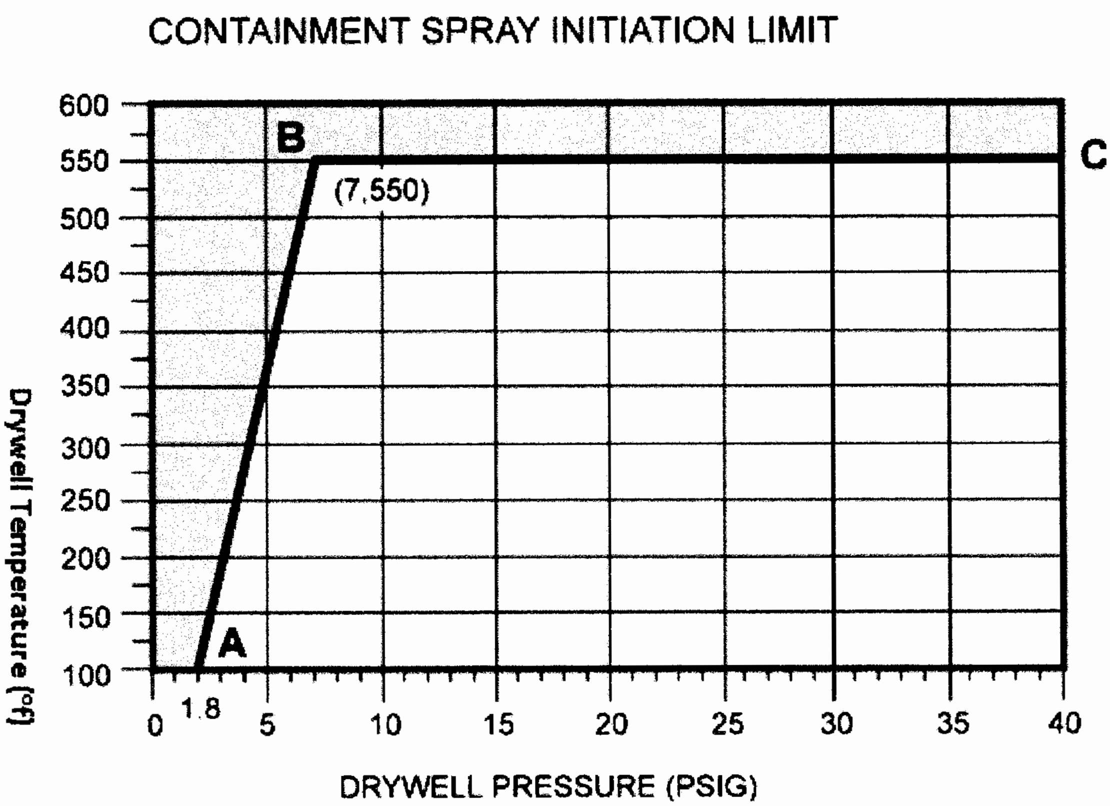
EXAMINATION ANSWER KEY

2016 SRO NRC Test

77 **ID: 1248442** **Points: 1.00**

A steam leak in the drywell has resulted in the following Containment parameters over the past four minutes:

	08:01	08:02	08:03	08:04
Drywell Pressure (psig)	2.9	3.1	4.5	4.8
Torus Pressure (psig)	2.0	2.2	3.0	3.5
Drywell Temperature (°F)	225	250	265	302



Which one of the following is the EARLIEST TIME at which Containment Spray can be initiated and maintain Primary Containment integrity in accordance with EMG-3200.02, Primary Containment Control?

A. 08:01

B. 08:02

C. 08:03

D. 08:04

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Answer: C

Answer Explanation				
K&A	295028 - High Drywell Temperature EA2.04 - Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Drywell pressure (4.2)			
Level: SRO	**Tier:** 1		**Group:** 1	
General References	EMG-3200.02			
Explanation	Proposed Answer: C Explanation: At 265 F, drywell pressure must be above 3.8 psig to spray. This requirement is met for these conditions. **Note:** K/A matches because question requires interpreting the relationship between high drywell temperature and the high drywell pressure and determining the mitigation strategy required for these conditions. A. Plausible if the applicant does not read and interpret the curve correctly. At 225F, drywell pressure must be above 3.2 psig to spray. This requirement is NOT met for these conditions. B. Plausible if the applicant does not read and interpret the curve correctly. At 250F, drywell pressure must be above 3.5 psig to spray. This requirement is NOT met for these conditions. D. Plausible if the applicant does not read and interpret the curve correctly. At 302F, drywell pressure must be above 4.1 psig to spray. This requirement is met for these conditions, however the "okay to spray" region was entered earlier. Also plausible if the applicant thinks they must wait until drywell temperature exceeds 300F.			
Lesson Plan **Learning Objective/**	2621.845.0.02- PRIMARY CONTAINMENT CONTROL LP PCC-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status			
References Provided	**ILT:** None		**LORT:** Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	**Memory or Fundamental Knowledge**		**Comprehension or Analysis**	X
10CFR55 Content	**55.41b**		**55.43b**	5
10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295028	**PRA:**	No	
Safety Function(s):	10	☒ **ILT**		
Category(s) (LORT Only):	N/A	☐ **LORT**		

78 **ID: 1250436** **Points: 1.00**

The reactor was at rated power when the turbine tripped. The following conditions exist:

- RPV water level has lowered to 124" and is recovering.
- RPV pressure is 960 psig and rising slowly.
- 20 control rods indicate position 06 and greater.
- All APRM/LPRM DNSCL lights are LIT.
- All electrical power busses are aligned as expected.
- All SCRAM SOLENOID GROUP lights are OFF.

Which of the following states the required action?

A. Trip all Reactor Recirculation Pumps, IAW RPV Control – With ATWS.

B. Control RPV water level 138" – 175", IAW RPV Control – With ATWS.

C. Vent the scram air header, IAW Support Procedure 21, Alternate Insertion of Control Rods.

D. Terminate and prevent injection into the RPV by all systems except CRD and boron injection systems IAW Support Procedure 17.

Answer: B

Answer Explanation

K&A	295037 - SCRAM Condition Present and Reactor Power Above APRM Downscale EA2.02 - Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Reactor water level (4.2)		
Level: SRO	Tier: 1		Group: 1
General References	EOP Users Guide		

Explanation	Proposed Answer: B Explanation: IAW with RPV control with ATWS the applicant must determine status of reactor, Type of ATWS to answer the question correctly. With the given information in the stem it can be determined that a hydraulic ATWS by the scram solenoid indicating off and also determine that Rx power is less than 2% by the APRM/LPRM downscale lights being lit. With this information Rx water level must be controll;ed at 138"-175" IAW RPV control with ATWS. **Note:** This question meets the SRO-only question guidelines for 10CFR55.43(b)(5) based on testing the ability to assess a plant conditions to prescribe the correct procedure section (in this case, choose the correct level control strategy in accordance with EOPs). A. It is plausible if the applicant does not recall that recirc pumps do not have to be tripped if power is determined to be less than 2%. During a ATWS with Rx power greater than 2% recirc are required to be tripped IAW RPV control with ATWS but since the APRM/LPRM downscale lights are lit this indicates that Reactor Power is less than 2% and there is no requirement to trip all Recirc pumps. C. It is plausible if the applicant improperly determines that there is an electric ATWS and therefore venting the scram header would be correct distractor. Since the scram solenoid lights are off venting the scram air header would have no effect on an hydraulic ATWS and therefore is not a required action for these conditions. D. It is plausible if the applicant does not recall that Rx power is still greater than 2% therefore performing SP-17 would be required. Since the APRM/LRPM downscale lights are lit this is indication that power is less than 2% and therefore SP-17 is not required IAW RPV control with ATWS.			
Lesson Plan Learning Objective/	2621.845.0.01B - RPV CONTROL-WITH ATWS EWA-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.			
References Provided	ILT: None		LORT: Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b		55.43b	5
10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.			

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295037	PRA:	No	
Safety Function(s):	10	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

79 **ID: 1248444** **Points: 1.00**

The plant is operating at 100% power when Chemistry reports the following data from their weekly isotopic reactor water sample surveillance:

- 10.55 μCi/gm Total Iodine
- 4.23 μCi/gm Dose Equivalent Iodine-131

Which one of the following is required and why?

A. Place the plant in SHUTDOWN within 12 hours, to ensure off-site release rates will NOT exceed limits if post-LOCA venting of the Drywell is required.

B. Restore reactor coolant activity to below the limit within 48 hours or place the plant in SHUTDOWN within 12 hours, to ensure off-site release rates will NOT exceed limits following a Main Steam Line Break.

C. Place the plant in SHUTDOWN within 12 hours, to ensure off-site release rates will NOT exceed limits following a Main Steam Line Break.

D. Restore reactor coolant activity to below the limit within 48 hours or place the plant in SHUTDOWN within 12 hours, to ensure off-site release rates will NOT exceed limits if post-LOCA venting of the Drywell is required.

Answer: C

Answer Explanation

K&A	295038 - High Off-Site Release Rate 2.1.25 - Ability to interpret reference materials, such as graphs, curves, tables, etc. (4.2)		
Level: SRO	Tier: 1		Group: 1
General References	TS 3.6.A		
Explanation	Proposed Answer: C Explanation: with reactor coolant >4 uCi/gm Dose Equivalent Iodine-131 TS 3.6.A.2 requires the reactor to be in at least shutdown condition within 12 hours. The basis for the TS is to limit the off-site release rate below 10CRD100 limits in the event of a Main Steam Line Break accident outside of primary containment. A. Plausible since this is the correct action for the conditions but is not the correct basis. If the applicant believes that since release rates would be higher during post-loca venting this would be the basis of the tech spec. B. Plausible if the applicant assumes that since activity in D.E.I-131 is greater than .2uCi/gm then operation may continue for up to 48 hours as stated in 3.6.A.1 and does not interpret 3.6.A.2 correctly then a shutdown would be required within 12 hours after the 48 hours. However 3.6.A.2 does not allow the 48 hours of operation if D.E.I-131 is greater than 4uCi/gm and a shutdown condition is required within 12 hours. D Plausible if the applicant assumes that since activity in D.E.I-131 is greater than .2uCi/gm then operation may continue for up to 48 hours as stated in 3.6.A.1 and does not interpret 3.6.A.2 correctly then a shutdown would be required within 12 hours after the 48		

	hours. However 3.6.A.2 does not allow the 48 hours of operation if D.E.I-131 is greater than 4uCi/gm and a shutdown condition is required within 12 hours. The basis is correct KA match justification: The applicant will need to interpret the provided reference in order to answer the question correctly.			
Lesson Plan Learning Objective/	2621.845.0.01B - RPV CONTROL-WITH ATWS EWA-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status.			
References Provided	ILT: TS 3.6.A with "objective" blacked out and without bases			LORT: Open
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge	X	Comprehension or Analysis	
10CFR55 Content	55.41b		55.43b	2
10CFR55 Explanation	Facility operating limitations in the technical specifications and their bases			
Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295038	PRA:	No	
Safety Function(s):	10	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

80 **ID: 1248445** **Points: 1.00**

The plant is operating at 35% power with the turbine online.

Which one of the following annunciators and VALIDATED indications would require a manual Reactor scram to be directed, in accordance with the associated Alarm Response Procedures?

A. Q-3-b, TURBINE MECH VIBRATION HI, alarms with turbine bearing vibration at 7 mils and stable.

B. R-5-c, GENERATOR STATOR TEMP HI, alarms with Generator gas temperature at 55°C and rising.

C. R-6-c, GENERATOR STATOR CLG TROUBLE, alarms with Generator temperature at 86°C and stable.

D. Q-4-b, SHELL ROTOR DIFF EXP HI/LO, alarms with HP turbine shaft-shell differential expansion indicated at 410 mils and rising.

Answer: D

Answer Explanation

K&A	295006 - SCRAM 2.4.31 - Knowledge of annunciator alarms, indications, or response procedures. (4.1)		
Level: SRO	Tier: 1		Group: 1
General References	RAP R-4-b	R-5-c, R-6-c,	Q-3-b

EXAMINATION ANSWER KEY

2016 SRO NRC Test

<table>
<tr><td>Explanation</td><td colspan="4">Proprosed Asnwer: D

Explaination: Q-4-b, shell rotor diff exp hi/lo, alarms at <100 mils or >400 mils and requires a manual scram if power is >30% and the alarm is validated. It's tied to 10CFR 55.43(b)5, which is, "Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations." So in the justification, make a statement that says, "This question meets the requirements of 10CFR 55.43(b)5 by requiring the applicant to assess various plant conditions and based on the assessment determine the appropriate procedure requiring a manual scram to be inserted."

A. Plausible – With reactor power above 30%, RAP Q-3-b requires a reactor scram when a turbine trip is required. The applicant needs to know the setpoint requiring a turbine trip. The lowest vibration setpoint requiring a turbine trip is 8 mils. Even though reactor power is above 30%, since a turbine trip is not required, neither is a scram.
B. Plausible – RAP R-5-c requires entry into ABN-11, Loss of Generator Stator Cooling, for the given conditions. ABN-11 requires a reactor scram if a turbine runback occurring. Generator gas temperature at 55°C and rising would bring in the alarm, but would not initiate a turbine runback. Therefore, a reactor scram is not required.
C. Plausible – RAP R-6-c will be in alarm with Generator temperature at 86°C. With reactor power at 35%, a reactor scram would be required if either 1) a Main Turbine runback was in progress, or 2) stator temperatures were rising. The applicant needs to know that Generator temperature at 86°C will not generate a runback.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621. 828.0.0050 - TURBINE AND TURBINE AUXILIARIES

MTA-10449- State the function and interpretation of system alarms (Main Turbine), alone and in combination, as applicable in accordance with the system RAPS.</td></tr>
<tr><td>References Provided</td><td colspan="2">None</td><td></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">New</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td>X</td><td>Comprehension or Analysis</td><td></td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td></td><td>55.43b</td><td>5</td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.</td></tr>
</table>

Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295006	PRA:	No	
Safety Function(s):	11	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

81 **ID: 1248446** **Points: 1.00**

A plant shutdown was in progress in preparation for a refuel outage. Current plant conditions are as follows:

- Shutdown Cooling Pump A is in service
- Shutdown Cooling Pump B is in service
- Shutdown Cooling Pump C is in standby
- RPV coolant temperature is 325 °F and lowering

The following annunciator just alarmed:

- DC-1 PWR LOST

The Operator reports that position indication to V-17-1 and V-17-2 have been lost (SDC Loop A suction valve and SDC Loop B suction valve).

Which of the following states the impact on the Shutdown Cooling System and the action required related to the Shutdown Cooling System **ONLY**?

	Impact on Shutdown Cooling	SDC Required Action
A.	The Shutdown Cooling System shall be declared inoperable	Isolate the Shutdown Cooling System **WITHIN** 4 hours
B.	Shutdown Cooling Loops A and B **ONLY** shall be declared inoperable	Remove Shutdown Cooling Loops A and B from service **WITHIN** 4 hours
C.	Declare impacted Shutdown Cooling System Primary Containment Isolation Valves inoperable	Restore Shutdown Cooling Primary Containment Isolation Valves to operable **BY THE TIME** the REACTOR MODE SELECTOR switch is placed in RUN on plant startup
D.	Declare impacted Shutdown Cooling System Primary Containment Isolation Valves inoperable	Restore Shutdown Cooling Primary Containment Isolation Valves to operable **PRIOR TO** declaring the reactor critical on plant startup

Answer: D

Answer Explanation			
K&A	295021 - Loss of Shutdown Cooling 2.1.32 - Ability to explain and apply system limits and precautions. (4.0)		
Level: SRO	Tier: 1		Group: 1
General References	305, RAP 9XF4d	UFSAR Table 6.2-12	T.S. 3.5.A.3

Explanation	Proposed Answer: D Explanation: The question stem shows that Shutdown Cooling (SDC) is in service with Loops A and B. IAW Procedure 305, Shutdown Cooling System Operation Precautions and Limitations (step 3.2.2) and the USAR reference, the SDC Loop suction and discharge valves are considered primary containment isolation valves. All of these 6 valves (suction & discharge for each of 3 loops) are powered from 125 VDC MCC DC1. Therefore, 4 of the 6 inoperable valves are open with RPV coolant temperature above 212 °F. These valves shall be declared inoperable. TS 3.5.A.3.a.(3) allows inoperable SDC containment isolation valves with RPV coolant temperature < 350 °F. The same Tech Spec requires that the inoperable valves be made operable prior to placing the reactor in the condition where Primary Containment is required (as when the plant is started-up). Additionally, from TS 3.5.A.3, primary containment shall be maintained when the reactor is critical or RPV temperature is above 212 °F. Therefore, there is no requirement to alter the current SDC configuration, although the valves are inoperable. But, the valves must be made operable prior to either declaring the reactor critical, or exceeding cold shutdown temperatures (ie, > 212 °F) [since either of these conditions require primary containment integrity]. A. Plausible if the applicant misinterprets Tech Spec requirements. There is no requirement to remove SDC from service. B. Plausible if the applicant misinterprets Tech Spec requirements. There is no requirement to remove SDC from service. C. Plausible – The SDC PCI valves do need to be declared inoperable. Hoever, since the reactor is past initial criticality and RPV coolant temperature is in excess of 500 °F when the reactor mode switch is placed in RUN (ie, this is past the 2 conditions that require primary containment to be established), verifying containment isolation valve operability at this point would be too late.				
Lesson Plan Learning Objective/	2621.845.0.0045 - SHUTDOWN COOLING SYSTEM SDC-10441 - Given the system logic/electrical drawings, describe the system trip signals, setpoints and expected system response including power loss or failed components.			
References Provided	T.S 3.5			LORT: Open
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b		55.43b	2

10CFR55 Explanation	Facility operating limitations in the technical specifications and their bases			
Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295021	PRA:	No	
Safety Function(s):	11	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

82 **ID: 1248447** **Points: 1.00**

The plant was at rated power when an event resulted in a **TOTAL** loss of instrument air and Electrical ATWS.

One hour later, the plant conditions include the following:

- All rods indicate a GREEN-GREEN backlight
- RPV pressure is 680 psig and steady
- RPV water level is 100" and is rising slowly
- Torus water level is 168" and rising slowly
- Torus water temperature is 161°F and rising slowly
- Torus pressure is 25 psig and steady

Which **ONE** of the following actions is required at this time? **(See Attached)**

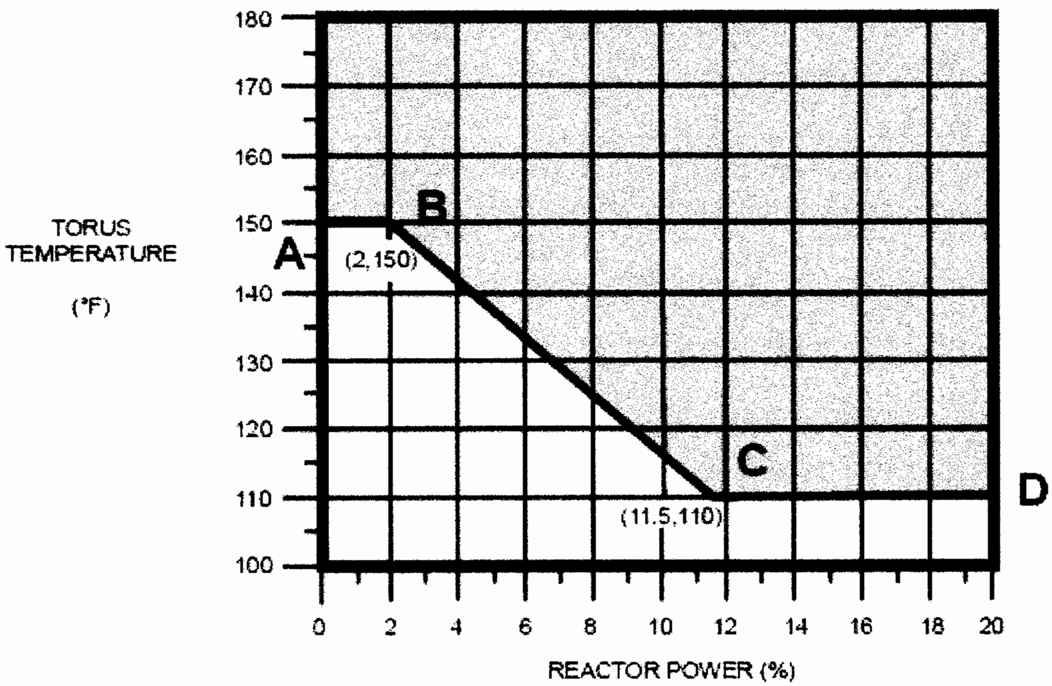
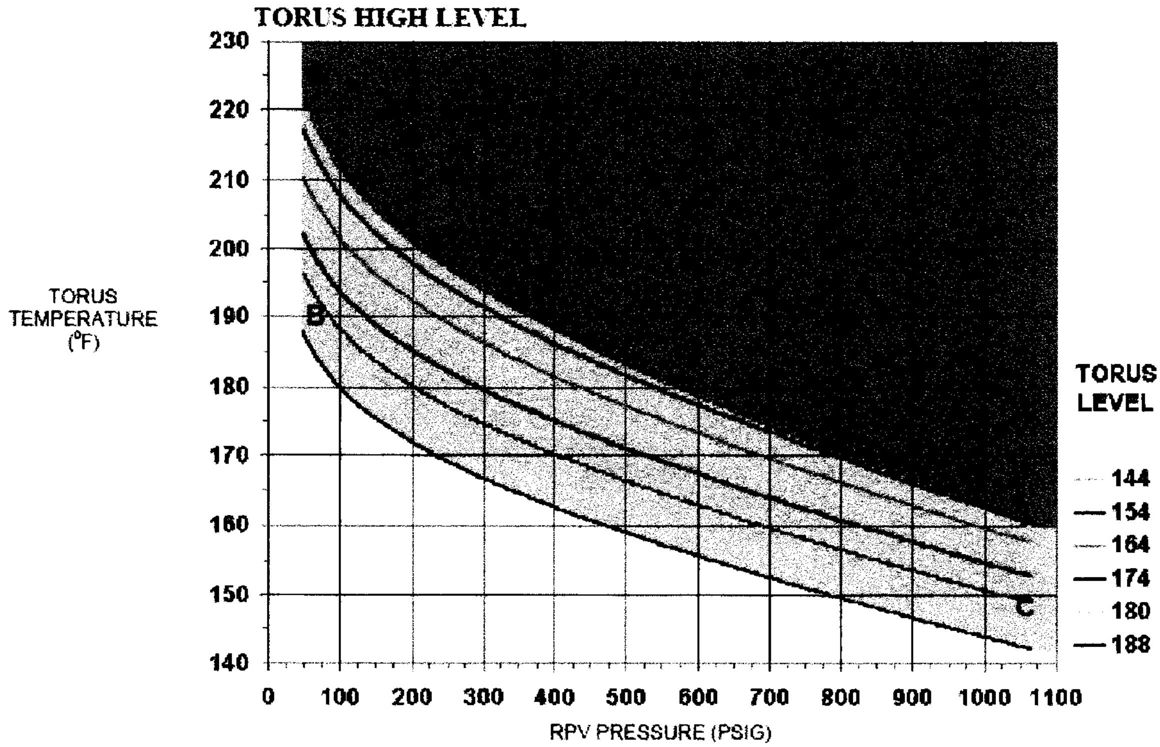


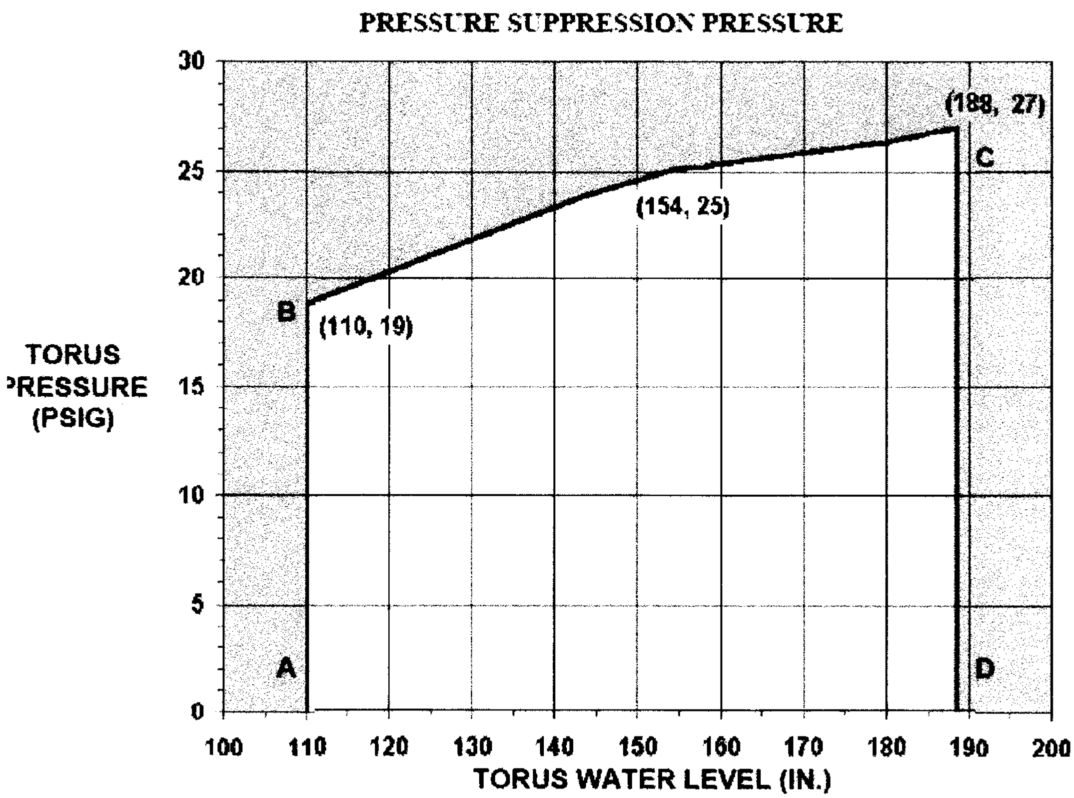
FIG. L BIIT

BORON INJECTION

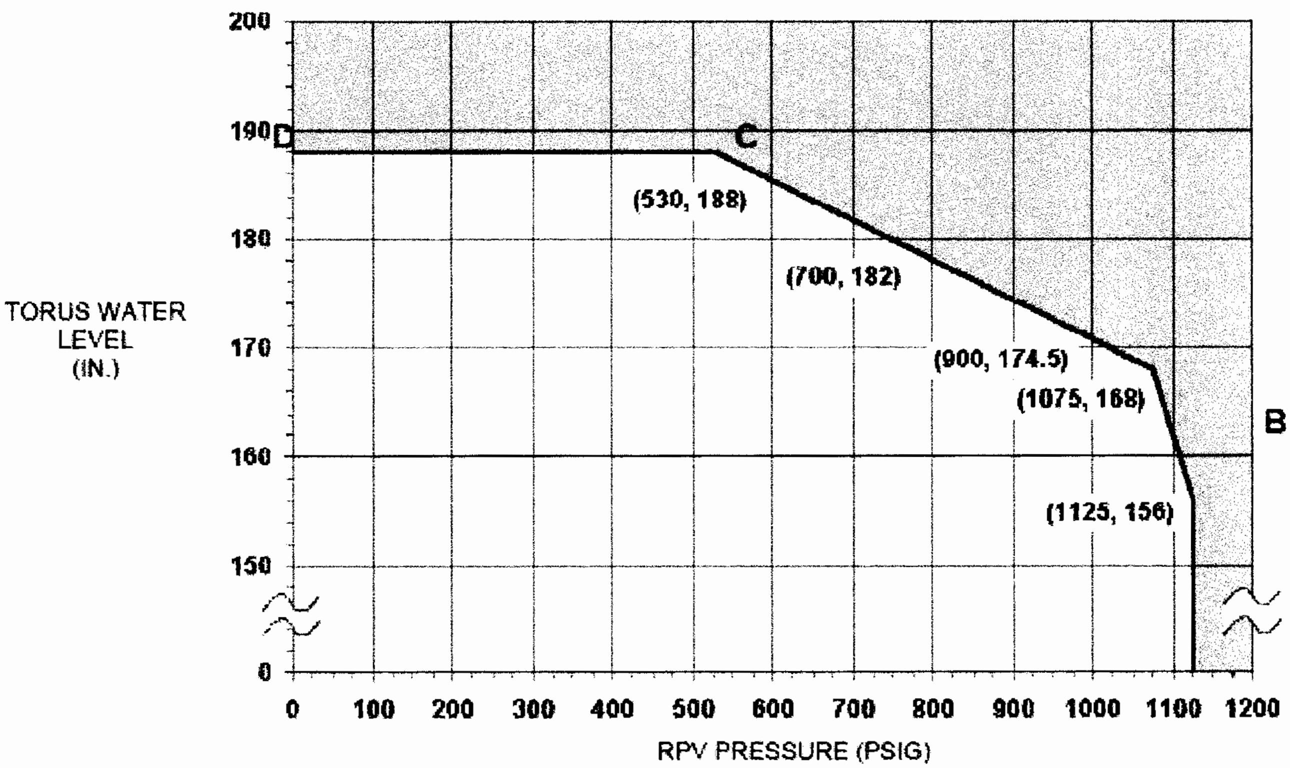
INITIATION TEMPERATURE



Heat Capacity Temperature Limit



TORUS LOAD LIMIT



A. Emergency Depressurize due to exceeding the BIIT.

B. Emergency Depressurize due to exceeding the PSP.

C. Lower RPV pressure with ICs to prevent exceeding the HCTL.

D. Lower RPV pressure with TBVs to prevent exceeding the TLL.

Answer: C

Answer Explanation			
K&A	295026 - Suppression Pool High Water Temperature EA2.01 - Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Suppression pool water temperature (4.2)		
Level: SRO	Tier: 1		Group: 1
General References	RPVC – No ATWS EOP	PCC EOP	EOP Users Guide

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Explanation	Proposed Answer: C Explanation: The question stem initially shows a high powered ATWS with a loss of instrument air. All control rods have since been fully inserted and RPV Control – No ATWS has been entered (and PCC EOP). The combination of a Torus water temperature is 161 °F and a Torus water level 168" places the plant close to, but not yet exceeding the heat capacity Temperature Limit (HCTL). With both parameters slowly rising, the point continues to get closer to exceeding HCTL. HCTL is mentioned in both the PCC EOP and RPVC – No ATWS EOP. In the pressure leg of RPVC – No ATWS EOP, it states that if Torus water temperature cannot be maintained below HCTL (and it is given that it is rising), then it directs to maintain RPV pressure below HCTL. Lowering RPV pressure with the Isolation Condensers can thus be used to lower RPV pressure. A. Plausible – The BIIT has in fact been exceeded. The applicant might not recall the correct action. Exceeding most graphs in the PCC EOP does require ED, the exceeding the BIIT is not one of them. B. Plausible – A Torus temperature of 161F and Torus pressure of 25 psig is right near the point where an ED would be required. The applicant may not read/interpret the PSP curve accurately. D. Plausible – It is true that RPV pressure and Torus level are at the point where lowering RPV Pressure is a correct action. With a total loss of instrument air, the MSIVs have closed, and the bypass valves are not available for pressure control.			
Lesson Plan Learning Objective/	2621.845.0.0056 - PRIMARY CONTAINMENT CONTROL LP PCC-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status			
References Provided	None			LORT: Open
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b		55.43b	5
10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.			
Justification for LORT questions with K/A values < 3.0	N/A			

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295026	PRA:	No	
Safety Function(s):	10	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

83 **ID: 1248448** **Points: 1.00**

A loss of coolant accident has occurred and the following conditions exist:

Drywell H2 concentration is 2.3%
Torus H2 concentration is 2.6%
Drywell O2 concentration is 2.4%
Torus O2 concentration is 2.3%

Which one of the following describes the relation to **(1)** the H2/O2 limit and **(2)** the required action in accordance with EOP Primary Containment Control?

	(1)	(2)
A.	Below the limit	Continue to sample the Drywell and Torus for H2 and O2.
B.	Below the limit	Direct Chemistry to sample the containment for radioactivity.
C.	Above the limit	Exit all EOPs and enter the Severe Accident Management Guidelines.
D.	Above the limit	Isolate the Primary Containment Vent and Purge valves being used for Primary Containment Pressure Control.

Answer: C

Answer Explanation			
K&A	500000 - High Containment Hydrogen Concentration. EA2.04 - Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Combustible limits for wetwell (3.3)		
Level: SRO	Tier: 1		Group: 2
General References	EOP Primary Containment Control	EOP User's Guide	

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Explanation	Proposed Answer: C Explanation: The limit in the Combustible gas leg of EOP Primary Containment Control is 2.5% H2 in the Drywell or the Torus. At 2.6% H2 in the Torus, this limit is exceed and requires Primary Containment Flooding in accordance with the SAMGs and all EOPs are exited. A. Plausible – Three of the four parameters are below the 2.5% limit. The action is required if the limit is not exceeded. B. Plausible – Three of the four parameters are below the 2.5% limit. This action is required if the limit is not yet exceeded and offsite release rates are expected to rise above UE level. D. Plausible – The limit has been exceeded for Torus H2. This action is required if the limit is not yet exceeded and offsite release rates have risen above UE level.			
Lesson Plan Learning Objective/	2621.845.0.0056 - PRIMARY CONTAINMENT CONTROL LP PCC-10445 - Given a set of system indications or data, evaluate and interpret them to determine limits, trends and system status			
References Provided	None		LORT: Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b		55.43b	5
10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.			
Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	500000	PRA:	No	
Safety Function(s):	10	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

84 **ID: 1248449** **Points: 1.00**

The reactor was at rated power when a LOCA occurred. Plant conditions include the following:

- Reactor has been scrammed and all rods at "00"
- RPV pressure is 159 psig and stable with 1 Condensate Pump still injecting
- RPV water level was just raised to 60" TAF, and is rising slowly
- Core Spray injection has been terminated
- Torus water level is 130" and rising at 1" per minute

Assume the Torus water level trend remains constant.

Which one of the following describes:

(1) When Torus water level will reach the Technical Specification Limit, and

(2) the action required in accordance with Technical Specifications?

	(1)	(2)
A.	<25 minutes	The reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.
B.	<25 minutes	Torus level shall be reduced to below the limit within 24 hours, or THEN the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.
C.	>25 minutes	The reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.
D.	>25 minutes	Torus level shall be reduced to below the limit within 24 hours, or THEN the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

Answer: A

Answer Explanation			
K&A	295029 - High Suppression Pool Water Level 2.2.42 - Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (4.6)		
Level: SRO	Tier: 1		Group: 2
General References	TS 3.5.A.1	EOP Users guide	

<table>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: A

Explanation: Tech Spec 3.5.A.1 states that any time the reactor is pressurized above atmospheric, the maximum Torus water volume is limited to 92,000 ft^3. The EOP bases state that 92,000 ft^3 correlates to 154" Torus water level. With Torus water level rising at 1" per minute and having to rise 24" (154"-130"), level would reach the limit in 24 minutes (<25 minutes). Tech Spec 3.5.A.1 states that if the limit is exceeded, the reactor shall be placed in the COLD SHUTDOWN CONDITION within 24 hours.

B. Plausible – <25 minutes is correct. However, 24 hours to fix the condition applies to Torus water temperature in the same technical specification, not Torus water level.
C. Plausible – >25 minutes is incorrect, but plausible if the applicant is not familiar with the relationship of torus water volume to level. The action is correct.
D. Plausible – >25 minutes is incorrect, but plausible if the applicant is not familiar with the relationship of torus water volume to level. 24 hours to fix the condition applies to Torus water temperature in the same technical specification, not Torus water level.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.828.0.0032 - PRIMARY CONTAINMENT

PCS-00422 - Referencing plant Technical Specifications (* from memory for Initial Candidates) and given a set of plant conditions, determine, as applicable, the: LCO Action Requirements (SRO ONLY)</td></tr>
<tr><td>References Provided</td><td colspan="2">None</td><td></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">New</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td></td><td>55.43b</td><td>5</td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.</td></tr>
<tr><td>Justification for LORT questions with K/A values < 3.0</td><td colspan="4">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="4">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="4">1</td></tr>
<tr><td>System ID No.:</td><td>295029</td><td>PRA:</td><td>No</td><td></td></tr>
</table>

Safety Function(s):	10	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

85 **ID: 1265031** **Points: 1.00**

The plant was at rated power with the following:

- An EMRV spuriously opened and was closed causing Torus temperature to rise to 96°F
- Containment Spray system 2 was placed in Torus cooling mode.
- ESW pump 52C is running.
- Annunciator RAP-B5b, TUBE/SHELL D/P LO.
- An operator reports from the field that a leak has developed from the discharge flange of ESW pump 52C.
- PPC indicates ESW flow at 2000 gpm and steady.

Which one of the following describes (1) the operability of ESW pump 52C and (2) the initial action and (3) initial Tech Spec requirement (if any)?

	(1)	(2)	(3)
A.	Operable	Start only ESW pump 52D	No Tech Spec action.
B.	Inoperable	Start only ESW pump 52A	Return the inoperable Pump to service within 7 days.
C.	Inoperable	Start only ESW pump 52D	Return the inoperable Pump to service within 15 days.
D.	Operable	Start only ESW pump 52A	No tech spec action.

Answer: C

Answer Explanation

K&A	295026 – Suppression pool high water temperature 2.2.37 - Ability to determine operability and/or availability of safety related equipment. (4.6)		
Level: SRO	Tier: 1		Group: 2
General References	310	RAP-B5b	

Explanation	Proposed answer: C Explanation: ESW pump 52C is inoperable due flow being below 3100 gpm. The only manual isolation valve for the pump is a common discharge isolation valve for 52C and 52D but there is an isolation check valve that would prevent flow from short cycling if ESW pump 52D was started therefore only ESW pump 52C is inoperable and the loop is still operable. Starting ESW pump 52D would be allowed as a corresponding action to drop torus temperature as procedures require starting alternate pumps as required. Per Tech Specs with only 52C pump being inoperable and loop still being operable the pump would have to be returned to service within 15 days. A is Plausible: If the applicant does not recall the flow requirement for an ESW pump to be considered inoperable and believes it is still operable then there would be no tech spec action and this would be a correct corresponding action as starting 52D would be the next step to raise flow in System 2 and clear the lo D/P alarm . Since the pump is inoperable then A is incorrect. B is Plausible: ESW pump 52C is inoperable due to flow being below 3100 gpm. Therefore it is correct to call the pump inoperable and if the applicant recalls that since there is no manual isolation valve at the discharge of the 52C then 52D would not be available therefore System 2 Loop would be inoperable and if the operator believes that 52A is part of same system as it is with Core Spray then starting 52A would provide the cooling necessary to the heat exchanger but since 52A is in system 1 and 52C is in system 2 for Containment Spray, starting 52A would not help cooldown the torus as its associated containment spray pumps are not running. Per Tech specs with a Containment Spray/ESW loop inoperable the loop would have to be returned within 7 days. But since there is a check valve at the discharge of 52C then 52D could still be started and therefore the loop would not be inoperable. B in incorrect since system 2 loop is not inoperable the 7 day LCO is not correct. D is Plausible: ESW pump 52C is inoperable due to flow being below 3100 gpm. If the applicant does not recall the flow requirement for an ESW pump to be considered inoperable and believes it is still operable then there would be no tech spec action. Also if the operator believes that 52A is part of same system as it is with Core Spray then starting 52A would provide the cooling necessary to the heat exchanger but since 52A is in system 1 and 52C is in system 2 for containment spray, starting 52A would not help cooldown the torus as its associated containment spray pumps are not running.. Since there is a check valve in the system pump 52D can be started to restore flow in system 2 to cool the torus. D is incorrect because 52C is inoperable.

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Lesson Plan Learning Objective/	2621.828.0.0009 - CONTAINMENT SPRAY/ESW SYSTEMS CNS-10451 - Referencing plant Technical Specifications (* from memory for Initial Candidates) and given a set of plant conditions, determine, as applicable, the: LCO Action Requirements **(SRO ONLY)**			
References Provided	None			LORT: Open
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b		55.43b	5
10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.			
Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	295026	PRA:	No	
Safety Function(s):	5	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

EXAMINATION ANSWER KEY

2016 SRO NRC Test

86 **ID: 1248451** **Points: 1.00**

The plant was in cold shutdown and was cooling down with the Shutdown Cooling System (SDC). An event occurred causing the following conditions to currently exist:

RECIRC PUMP SUCTION TEMPS indicates 215 °F
The Primary Containment is still inerted
RPV water level is 175" and steady

Another event then occurs as shown in the timeline below:

0800 Annunciator RBCCW – SURGE TANK LVL HI/LO alarms
0804 The EO reports the RBCCW Surge Tank indicates 1" and lowering and the Tank makeup valve is full open
0806 The Radwaste Operator reports RB Floor Drain Sump 1-7 high level is in alarm
0808 Maintenance reports that they are unable to repair the leak
0809 The SM observes Drywell pressure at 1.7 psig and steady and Drywell temperature at 155 °F and steady
0810 The SM starts the 1-hour clock to monitor entry into EAL MA5(1)

Which of the following shall the SRO direct **NEXT**?

A. Operate all available Drywell Coolers, IAW SP-27, Maximizing Drywell Cooling

B. Confirm Primary Containment Isolation IAW the Primary Containment Contol EOP

C. Isolate the Reactor Water Cleanup System IAW the Secondary Containment Control EOP

D. Initiate Isolation Condensers by placing the Condensate Return DC valves to OPEN IAW 307, Isolation Condenser System

Answer: A

Answer Explanation

K&A	400000 - Component Cooling Water System A2.02 - Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: High/low surge tank level (3.0)		
Level: SRO	Tier: 2		Group: 1
General References	EP-AA-1010	ABN-3	ABN-19

EXAMINATION ANSWER KEY

2016 SRO NRC Test

<table>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: A

Explanation: The plant is > 212 °F and cooling down with SDC with all 3 SDC pumps in service. Then, indications are provided which show an unisolable leak in RBCCW (lowering surge tank level and high level in the floor drain tank, and not corrected quickly). Operation of the Drywell coolers IAW SP-27 is directed from the Primary Containment Control EOP. Conditions show parameters greater than the entry conditions (DW temperature & pressure) for the EOP. Thus, the SP can be used to start all available DW fans to drop temperature.

B. Plausible – Since primary containment control EOP is entered then SP-1, confirmation of primary containment isolation is an available SP to use but since the parameters are below containment isolation signals it is not required to be used yet.

C. Plausible – With a loss of RBCCW, it is suggested that RWCU be removed from service. The RB floor drain sump 1-7 is an entry into the Secondary Containment Control EOP. In the Secondary Containment Control EOP, it directs isolation of leaking systems, which in this case, is RBCCW – not RWCU. Thus isolation/removal of RWCU is directed from the loss of RBCCW ABN and not the EOP.
D. Plausible – Since the RPV has lost its cooling medium and is heating up, Isolation Condensers can used now that RPV temperature is > 212 °F. But with RPV water level >160", initiation per the normal procedure is not allowed.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.828.0.0035 - RBCCW

RBC-10450 – Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation in accordance with applicable ABN, EOP and EOP support procedures, and EP procedures</td></tr>
<tr><td>References Provided</td><td>None</td><td colspan="2"></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">Modified</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td></td><td>55.43b</td><td>6</td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	400000	PRA:	No	
Safety Function(s):	8	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

87 **ID: 1248452** **Points: 1.00**

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

- ROD CONTROL – CONTROL AIR PRESS LO

The TB Operator reports that the in-service drying tower has isolated and the standby drying tower cannot be placed into operation. The SRO ordered a manual reactor scram when INSTR AIR SUPPLY PRESS indicated < 60 psig and lowering. With the REACTOR MODE SELECTOR switch in SHUTDOWN, the current plant conditions are as follows:

- **ALL** of the LPRM amber lights on the full core display are LIT
- RPV water level is 120" and rising
- The MASTER RECIRC SPEED CONTROLLER indicates 35 hertz
- 8 control rods indicate position 22

Assuming that a drying tower **CANNOT** be restored and indicated air pressure has decayed to 0 psig, which of the following states the plant impact and the required action?

	Plant Impact	Required Action
A.	Main steam flow to the turbine and/or condenser is isolated.	Stabilize RPV pressure below 1045 psig with the Isolation Condensers.
B.	The Recirculation MG fluid couplers have locked up.	Place the Recirculation Pumps in local manual control and reduce to minimum.
C.	The CRD DRIVE WATER Pressure Control valve has failed closed.	Place the bypass Pressure Control valve in-service and manually insert control rods.
D.	The Feedwater MFRVs have locked Up.	Terminate and prevent Feedwater by closing the Heater Bank Outlet valves.

Answer: A

Answer Explanation

K&A	300000 - Instrument Air System A2.01 - Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Air dryer and filter malfunctions (2.8)		
Level: SRO	Tier: 2		Group: 1
General References	EOP RPV Control with ATWS	ABN-35	RAP-H1a

<table>
<tr><td rowspan="2">Explanation</td><td colspan="4">Proposed Answer: A

Explanation: The question describes a loss of air event and a failure of the reactor to scram, with reactor power < 2% (since all LPRM amber lights are lit). With air pressure at 0 psig, the outside MSIVs have closed and thus steam flow to the turbine or condenser is isolated, and IAW the ATWS EOP, pressure control should be stabilized < 1045 psig. Pressure control with the Isolation Condensers is allowed (as long as RPV water level is < 160”, which it is).</td></tr>
<tr><td colspan="4">B. Plausible – It is true that with a loss of instrument air, the Recirculation MG fluid couplers (scoop tubes) lock up in their current position. The question stem shows that the recirculation pumps are currently at 35 hertz, which is way above the minimum. IAW the ATWS EOP, flowing back recirculation flow to minimum is required when the main generator is on-line. The stem does not provide any indications that the turbine generator did not trip, and thus it is correct to assume that it has. Since the generator is not online, reducing recirculation flow is not required (although the step is the correct way to control recirculation flow during a loss of air event)
C. Plausible – It is true that the in-service CRD FCV fails closed on loss of air, but the CRD drive water PCV is motor operated, and is unaffected by the loss of air. Since the CRD FCV has failed closed, CRD water supply is not available downstream to manually insert control rods.
D. Plausible – The feedwater MFRV will lock up on loss of air (but may slowly drift open or closed). But since RPV water level is 120” and reactor power < 2%, there is no need to terminate and prevent feedwater (although the listed method is one correct method to control feedwater flow during a loss of air event).</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.845.0.0026 - MAIN STEAM SYSTEM

MSS-10450 - Describe and interpret procedure sections and steps for plant emergency or off-normal conditions that involve this system including personnel allocation and equipment operation IAW applicable ABN, EOP & EOP support procedures and EP procedures.</td></tr>
<tr><td>References Provided</td><td>None</td><td colspan="2"></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">Bank</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td></td><td>55.43b</td><td>5</td></tr>
</table>

10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations			
Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	300000	PRA:	No	
Safety Function(s):	8	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

88 **ID: 1248453** **Points: 1.00**

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

- CNTRL DC – 1A2 DC LOST

Which of the following states the impact on the Core Spray System (Consider Active Components **ONLY**) and the **MOST LIMITING** Technical Specification action statement required from section 3.4 ?

	Core Spray System 1 Inoperable Components	Core Spray System 2 Inoperable Components	TS 3.4 Action Statement
A.	One Booster Pump **AND** One Main Pump	One Booster Pump **AND** One Main Pump	The reactor may remain in operation not to exceed 15 days
B.	One Booster Pump **ONLY**	One Booster Pump **ONLY**	The reactor may remain in operation not to exceed 7 days
C.	One Booster Pump **AND** One Main Pump	One Booster Pump **AND** One Main Pump	The reactor may remain in operation not to exceed 7 days
D.	One Booster Pump **ONLY**	One Booster Pump **ONLY**	The reactor may remain in operation not to exceed 15 days

Answer: B

Answer Explanation

K&A	209001 - Low Pressure Core Spray System 2.2.38 - Knowledge of conditions and limitations in the facility license. (4.5)		
Level: SRO	Tier: 2		Group: 1
General References	TS 3.7, TS 3.4	ABN-55	RAP-U3d

<table>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: B

Explanation: The annunciator in the stem describes a loss of DC control power to USS 1A2 (which powers a core spray booster pump in each Core Spray System. System 1 includes the A/C booster pumps and System 2 includes the B/D booster pumps. When DC power is lost, 1 booster in each core spray system is lost. therefore with two of the four redundant active loop components in the core spray systemn not in the same lop (sytm 1 or systm 2 are inoperable the reactor may remain in operation not to exceed 7 days. None of the (Parallel Isolation Valves (PIVs) are directly affected by the loss of DC control power to USS 1A2, and are all still functioning.

A. Plausible – With USS 1A2 control power lost, the plant must be be shutdown after 7 days. Due to CS still being able to operate at designed flowrate, even with a loss of one booster pump and one PIV in each system, the student may believe the plant is in a 15 day LCO from TS 3.4 requirements with one or two non-redundant CS components in each loop inoperable.
C. Plausible – The required action is correct. However, Core Spray Main Pumps are powered from 4160 VAC power supplies and are not affected.
D. Plausible – With USS 1A2 control power lost, the plant must be be shutdown after 7 days. Due to CS still being able to operate at designed flowrate, even with a loss of one booster pump and one PIV in each system, the student may believe the plant is in a 15 day LCO from TS 3.4 requirements with one or two non-redundant CS components in each loop inoperable.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.828.0.0010 - CORE SPRAY SYSTEM

CSS-10451 - Referencing plant Technical Specifications (* from memory for Initial Candidates) and given a set of plant conditions, determine, as applicable, the: LCO Action Requirements (SRO ONLY)</td></tr>
<tr><td>References Provided</td><td>T.S 3.7, 3.4</td><td colspan="2"></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">Bank</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td></td><td>55.43b</td><td>1</td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Conditions and Limitations in the facility license</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	209001	PRA:	No	
Safety Function(s):	4	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

89 **ID: 1248454** **Points: 1.00**

The following conditions exist:

- Reactor power is 0 percent
- Reactor water level is 84 inches and steady
- RPV pressure is 115 psig and lowering
- RPV temperature is 347°F
- CHRRMS is 700 R/HR and rising slowly
- RBO has reported steam in the SDC heat exchanger room
- SDC area rad monitors (IB06-E, 51' HX room (C HX)) are 800 MR/HR and rising slowly
- V-17-54 has stuck open
- Reactor Building D/P is +.15 " W.G.

What is the highest level of classification for the given conditions?

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: D

Answer Explanation			
K&A	205000 - Shutdown Cooling System (RHR Shutdown Cooling Mode) 2.4.41 - Knowledge of the emergency action level thresholds and classifications. (4.6)		
Level: SRO	Tier: 2		Group: 1
General References	EP-AA-1010		

EXAMINATION ANSWER KEY

2016 SRO NRC Test

<table>
<tr><td rowspan="1">Explanation</td><td colspan="5">Proposed Answer: D

Explanation: The applicant needs to recognize which fission product barrier is lost or potentially lost then determine what classification is required. With CHRRMS reading 700 R/HR, a loss of fuel clad and RX coolant system have occurred. Since RX Water level is below LoLo for a primary containment isolation signal and V-17-54 is stuck open, then a loss of containment has also occurred. Therefore with a loss of all 3 fission product barriers a General Emergency is the highest classification with the current conditions.

A. Plausible – If the applicant recognizes the loss of containment but not recognize CHRRMS, then an unusual event would be correct.
C. Plausible – If the applicant recognizes the loss of fuel clad or RX coolant system due to CHRRMS but does not recognize the loss of containment, then Alert would be correct.
D. Plausible – If the applicant recognizes the loss of fuel clad and RX coolant system but does not recognize the loss of containment, then a Site Area Emergency would be correct.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="5">2685.792.0.0010 - NEI 99-01 Rev 5 EALs

EPAA101001-05 - Identify correct EAL classification.</td></tr>
<tr><td>References Provided</td><td colspan="2">PAGEs OCGS 2-1 through 2-10 (EP-AA-1010)</td><td colspan="2"></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="5">New</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="5">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td colspan="2">Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td></td><td colspan="2">55.43b</td><td>5</td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="5">Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.</td></tr>
<tr><td>Justification for LORT questions with K/A values < 3.0</td><td colspan="5">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="5">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="5">1</td></tr>
<tr><td>System ID No.:</td><td>205000</td><td>PRA:</td><td colspan="2">No</td><td></td></tr>
<tr><td>Safety Function(s):</td><td>4</td><td>☒ ILT</td><td colspan="2"></td><td></td></tr>
<tr><td>Category(s) (LORT Only):</td><td>N/A</td><td>☐ LORT</td><td colspan="2"></td><td></td></tr>
</table>

90 **ID: 1248455** **Points: 1.00**

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

- An EMRV inadvertently opened and cannot be closed.
- Due to the stuck open EMRV, Torus temperature has risen to 95F and continues to slowly rise.

Which one of the following is the required action per technical specifications and the associated basis for the action?

	Tech Spec Action	Tech Spec Basis
A.	Be in COLD SHUTDOWN within 24 hours	Ensure that the maximum peak Torus temperature does not exceed 110°F if an ED was performed
B.	Be in COLD SHUTDOWN within 24 hours	Ensure that the maximum peak Torus temperature does not exceed 160°F if an ED was performed
C.	Be in COLD SHUTDOWN within 30 hours	Ensure that the maximum peak Torus temperature does not exceed 160°F if an ED was performed
D.	Be in COLD SHUTDOWN within 30 hours	Ensure that the maximum peak Torus temperature does not exceed 110°F if an ED was performed

Answer: B

Answer Explanation

K&A	239002 - Safety Relief Valves A2.03 - Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: - Stuck open SRV (4.2)		
Level: SRO	Tier: 2		Group: 1
General References	TS 3.5.A.1 and associated bases		

<table>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: B

Explanation: IAW TS 3.5.A.1, the Maximum Torus water temperature is 95F at power. TS 3.5.A.1.d states that if this limit is exceeded, be in the COLD SHUTDOWN condition within 24 hours. The basis for this action is to avoid excessive Torus loading following a depressurization using EMRVs. This is accomplished by ensuring Torus temperature does not exceed 160F following any period of EMRV operation.
TS 3.5 Bases state the following in regards to maximum Torus temperature: Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

<u>Note:</u> This question meets the KA by test the comprehensive portion (part 'b') of the KA statement.

A. Plausible – Cold Shutdown in 24 hours is correct. The value of 110F is plausible if the student confuses this with the maximum temperature allowed where a reactor scram is required.
C. Plausible – The normal shutdown LCO action statement to be in Cold Shutdown if one is not given is 30 hrs. However, the Torus temp tech specs gives a specific value of 24 hrs. 160F is the correct basis value.
D. Plausible – The normal shutdown LCO action statement to be in Cold Shutdown if one is not given is 30 hrs. However, the Torus temp tech specs gives a specific value of 24 hrs. The value of 110F is plausible if the student confuses this with the maximum temperature allowed where a reactor scram is required.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.845.0.0056 - Primary Containment,

PCC-422</td></tr>
<tr><td>References Provided</td><td colspan="2">None</td><td></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">Bank</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td>X</td><td>Comprehension or Analysis</td><td></td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td></td><td>55.43b</td><td>2</td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Facility operating limitations in the technical specifications and their bases</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	239002	PRA:	No	
Safety Function(s):	3	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

91 **ID: 1248456** **Points: 1.00**

A reactor startup is in progress following an extended outage with the following conditions present:

- All IRMs are on range 2
- 8 control rods have been withdrawn
- During withdrawal of the 9th control rod the following alarm is received:
    - ROD BLOCK
- The rod block is determined to be from a failure of the Rod Worth Minimizer (RWM)
- All attempts to restore the RWM fail.
- The RWM has been operable for all required conditions over the last 12 months.

Which one of the following describes the implications on the reactor startup in accordance with Technical Specifications?

The startup…

A. Must be terminated and all control rods must be reinserted in reverse order.

B. Must be placed on hold with the last known rod pattern maintained until the RWM can be restored.

C. May continue only if a second qualified individual and a reactor engineer are stationed to verify compliance with the approved rod withdrawal sequence.

D. May continue only with station senior management approval.

Answer: C

Answer Explanation

K&A	201006 - Rod Worth Minimizer System (Plant Specific) A2.07 - Ability to (a) predict the impacts of the following on the ROD WORTH MINIMIZER SYSTEM (RWH) (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: RWM hardware/software failure: P-Spec(Not-BWR6) (2.8)		
Level: SRO	Tier: 2		Group: 2
General References	TS 3.02.B.2		

EXAMINATION ANSWER KEY

2016 SRO NRC Test

<table>
<tr><td rowspan="1">Explanation</td><td colspan="4">Proposed Answer: C

Explanation: Per TS 3.02.B.2, If the RWM becomes inoperable prior to startup or prior to withdrawing the first 12 control rods, startup may continue provided a second licensed individual is available to ensure the rod program is maintained and within the previous 12 months, a startup has not been completed without the RWM operable.

A. Plausible – The rod worth minimizer is required to be operable until reactor power reaches 10% of rated power. Reinserting rods in the reverse order would ensure rod worth is minimized, however, that is not the direction per tech specs.
B. Plausible – The rod worth minimizer is required to be operable until reactor power reaches 10% of rated power. In the event is becomes inoperable under the given circumstances, it is not required to restore prior to proceeding with the startup. If the RWM had been inoperable when required within the previous 12 months, this would be a correct answer.
D. Plausible – The startup may continue. However, while station management approval may be required by other procedures, it is not required by tech specs, nor is it enough to allow the startup to recommence.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621. 828.0.0041 - ROD WORTH MINIMIZER

RWM-10451 -Referencing plant Technical Specifications (* from memory for Initial Candidates) and given a set of plant conditions, determine, as applicable, the: LCO Action Requirements (SRO ONLY)</td></tr>
<tr><td>References Provided</td><td>None</td><td colspan="2"></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">New</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td></td><td>55.43b</td><td>2</td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Facility operating limitations in the technical specifications and their bases</td></tr>
<tr><td>Justification for LORT questions with K/A values < 3.0</td><td colspan="4">N/A</td></tr>
<tr><td>Time to Complete:</td><td colspan="4">1-2 minutes</td></tr>
<tr><td>Point Value:</td><td colspan="4">1</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 SRO NRC Test

System ID No.:	201006	PRA:	No	
Safety Function(s):	7	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

92 **ID: 1248457** **Points: 1.00**

The plant is shutdown with refuel activities in-progress. The Control Room is notified water level in the spent fuel pool dropped 5ft and is now steady. A few minutes later, the following annunciators alarmed:

- AREA MON HI
- STACK EFFLUENT HI

The Operator reports the following area radiation monitors in alarm (assume radiation is greater than MAX NORMAL and less than MAX SAFE for the alarms listed below):

- SPENT FUEL POOL AREA, C-5
- FUEL POOL LOW RANGE, C-9

The Operator reports the high range stack monitor is reading 3.02 E-09 amps and lowering at time 12:00.

The Operator reports the high range stack monitor is now ready 4.00 E-10 amps and steady at time 12:05

Which is the HIGHEST emergency plan classification warranted at 12:10

A. Unusual Event

B. Alert

C. Site Area Emergency

D. General Emergency

Answer: A

Answer Explanation			
K&A	233000 - Fuel Pool Cooling and Clean-up 2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm. (4.3)		
Level: SRO	Tier: 2		Group: 2
General References	EP-AA-1010		

<table>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: A

Explanation: IAW EP-AA-1010, a UE is required under the given circumstances: RU2

1. a. VALID indication of uncontrolled drop in water level in the Reactor Cavity, Spent Fuel Pool or Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water as indicated by: e. Reactor Cavity water level < 583 inches. (GEMAC Wide Range, floodup calibration)
OR
f. Report of visual observation of an uncontrolled drop in water level in the Reactor Cavity or Spent Fuel Pool.
AND
b. UNPLANNED VALID Area Radiation Monitor reading rise on one or more radiation monitors in Table R2.
OR
2. UNPLANNED VALID Area Radiation Monitor readings rise by a factor of 1000 over NORMAL LEVELS or VALID upscale reading.
Since the high range stack monitor reading when from an SAE level down to below an alert level a Unusual Event is correct

B. Plausible – If the applicant believes that since the stack high range monitors are on scale and are steady then an Alert is applicable because IAW with the EAL's the Emergency director should declare the event as soon as it has been determined that the condition has exceeded or will likely exceed the applicable time. SInce the Stack high range readings have dropped below the threshold of an Alert this is not the correct choice for this condition.
C. Plausible – If the applicant believes that since the stack high range monitors have exceeded the values of an Alert and mis-interpretes the readings for an SAE then this would be an applicable choice. Since the stack readings lowered back down this would be a wrong interprepration of the EAL's.
D. Plausible – If the applicant mis-interperates the stack effluent readings to have exceeded the GE thereshold then GE level would be applicable.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2685. 792.0.0010 - NEI 99-01 Rev 5 EALs

EPAA101001-05 - Identify correct EAL classification.</td></tr>
<tr><td>References Provided</td><td>Page OCGS 2-11 & 2-12 out of OC-AA-1010</td><td></td><td colspan="2">LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">Bank</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 SRO NRC Test

10CFR55 Content	55.41b		55.43b	5
10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.			
Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	233000	PRA:	No	
Safety Function(s):	9	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

93 **ID: 1248458** **Points: 1.00**

The plant is operating at 100% power. I & C reports that a document review has revealed the Turbine Stop Valve closure **<u>scram setpoints</u>** are set according to the table below.

Turbine Stop Valve	Channel 1 Setpoint (% valve closure)	Channel 2 Setpoint (% valve closure)
TSV-1	9%	10%
TSV-2	8%	11%
TSV-3	9%	10%
TSV-4	8%	12%

Which one of the following describes the significance of these setpoints and their effects following a Turbine Trip in accordance with Technical Specifications?

The Channel...

A. 2 setpoints are too high. This will narrow the margin to MCPR following a Turbine Trip.

B. 1 setpoints are too low. This will narrow the margin to MCPR following a Turbine Trip.

C. 2 setpoints are too high. This will result in an excessive RPV level transient following a Turbine Trip

D. 1 setpoints are too low. This will result in an excessive RPV level transient following a Turbine Trip.

Answer: A

Answer Explanation

K&A	245000 - Main Turbine Generator and Auxiliary Systems 2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. (4.7)		
Level: SRO	Tier: 2		Group: 2
General References	TS 2.3	TS 3.1.1	

Explanation	Proposed Answer: A Explanation: Once the SRO evaluates plant performance data of the turbine stop valve set points he makes an operational judgment based on operating characteristics that channel 2 setpoint for TSV 2 and 4 are above 10% which is outside the requirements of TS 2.3.M and TS 3.1.1 of equal to or less than 10%. The turbine stop valve closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to the worst case transient of a load rejection and subsequent failure of the bypass. The scram setpoints are chosen to ensure MCPR is not violated during the transient. B. Plausible – MCPR is the concern. However, the closure scram setpoint must be within 10%, therefore less than 10% closure is ok. C. Plausible – The setpoints are too high. An RPV level transient will occur, but this is not the concern in tech specs. D. Plausible – The closure scram setpoint must be within 10%, therefore less than 10% closure is ok. An RPV level transient will occur, but this is not the concern in tech specs.			
Lesson Plan Learning Objective/	2621. 828.0.0050 - TURBINE AND TURBINE AUXILIARIES MTA-10452 - Identify and explain each surveillance required for this system (Main Turbine, Turbine Lube Oil) including personnel allocation and equipment operation.			
References Provided	none		LORT: Open	
Question Source (New, Modified, Bank)	New			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b		55.43b	2
10CFR55 Explanation	Facility operating limitations in the technical specifications and their bases			
Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	245000	PRA:	No	
Safety Function(s):	4	☒ ILT		

Category(s) (LORT Only):	N/A	☐ LORT		

94 **ID: 1248459** **Points: 1.00**

The plant is in a refuel outage and fuel movements are in-progress on the refuel floor.

Which of the following is/are the responsibilities of the Fuel Handling Director (SRO) on the Bridge, IAW procedure 205.0, Reactor Refueling?

1. Signing for completion of each move on the Fuel Move Sheet.
2. Turning off the Bridge power supply if the Bridge controls fail.
3. Directly supervising the manual movement of fuel and controls in the core.
4. Ensuring all license requirements for refueling are satisfied.

A. 1 **ONLY**

B. 4 **ONLY**

C. 1 and 3

D. 2 and 4

Answer: C

Answer Explanation

K&A	2.1.40 - Knowledge of refueling administrative requirements. (3.9)		
Level: SRO	Tier: 3		Group:
General References	205		

<table>
<tr><td>Explanation</td><td colspan="4">Proposed Answer: C

Explanation: IAW procedure 205, the FHD (SRO) is responsible for the following:
1) directly supervising all core alterations; 2) having no other concurrent duties during core alterations; 3) signing for completion of each move on the fuel move sheet; 4) maintaining proper communication with the control room licensed operator; 5) assuring proper execution of core alterations IAW procedures and the fuel bundle orientation map; 6) ensuring no other activities in/around the fuel pool and reactor cavity during refuel operations that could distract the bridge operators or create any physical interference with refuel equipment; and, 7) notify the SM and RE of any refuel errors.
Of those listed in the question, only selection 1 and 3 (Answer C) is required by procedure as the FHD responsibility. IAW TS 1.21, core alterations includes the manual movement of fuel and controls in the core.

A. Plausible – Selection 1 is correct. However, selection 3 is also a responsibility.
B. Plausible if the applicant does not recall specific responsibilities. Selection 4 is a responsibility of the Shift Manager NOT the fuel handling director.
D. Plausible if the applicant does not recall specific responsibilities. Selection 4 is a responsibility of the Shift Manager NOT the fuel handling director. Selection 2 is a responsibility of the Fuel Move Spotter NOT the fuel handling director</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.812.0.0003 - REFUELING

RFL-00323 - State the responsibilities of the following personnel during refueling operations IAW procedure 205.0: Fuel Handling Director (FHD)</td></tr>
<tr><td>References Provided</td><td colspan="2">none</td><td></td><td>LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">Bank</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td>X</td><td>Comprehension or Analysis</td><td></td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td></td><td>55.43b</td><td>7</td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Fuel handling facilities and procedures</td></tr>
<tr><td>Justification for LORT questions with K/A values < 3.0</td><td colspan="4">N/A</td></tr>
</table>

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	14	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

95 **ID: 1248460** **Points: 1.00**

The following conditions exist:

- Reactor head is being reinstalled following a Refuel Outage to press up and conduct the NSSS leak test.
- This has been classified as a High Risk Evolution due to being a Heavy Lift over irradiated fuel, in accordance with MA-AA-716-022, Control of Heavy Loads Program.

Which of the following items are specifically required?

(1) Shift Manager or designee's approval prior to the lift
(2) Secondary Containment Integrity is operable prior to lift and throughout the leak test
(3) Maintenance Manager must ensure diverse Safe Shutdown Equipment is available prior to the lift

A. (1) and (2) only

B. (2) and (3) only

C. (1) and (3) only

D. (1), (2), and (3)

Answer: A

Answer Explanation			
K&A	2.2.7 - Knowledge of the process for conducting special or infrequent tests. (3.6)		
Level: SRO	Tier: 3		Group:
General References	MA-AA-716-022		

Explanation	Proposed Answer: A Explanation: Heavy Lifts over irradiated fuel require SM or designee approval (step 3.3) and Secondary Containment to be operable (step 4.6.13). Shift Manager or his designee will ensure that any any required redundant safe shutdown equipment is available at the time of the lift (step 3.6) **Note:** This question meets the SRO-level guidelines because the High Risk Evolution and Heavy Load processes are part of a network of processes involved in dealing with operating changes in the facility and the question tests an SRO function within the process (risk assessment/management). B. Plausible – Secondary Containment must be operable. Ensuring safe shutdown equipment is available at the time of the lift is required but the Shift Manger or designee must ensure and it does not have to be only the SOS or Ops director. C. Plausible – Shift Manager or designee's approval is required. Ensuring safe shutdown equipment is available at the time of the lift is required but the Shift Manger or designee must ensure and it does not have to be only the SOS or Ops director D. Plausible – Shift Manager or designee's approval is required. Secondary Containment must be operable. Ensuring safe shutdown equipment is available at the time of the lift is required but the Shift Manger or designee must ensure and it does not have to be only the SOS or Ops director. It is plausible if the applicant believes that since the reistallation of the reactor head is a maintenance procedure that the maintenance manager must ensure safe shutdown equipment available since most of the operations procedures requires a manager level to ensure safe shutdown equipment available.				
Lesson Plan Learning Objective/	2621.828.0.0030 - NUCLEAR STEAM SUPPLY SYSTEM NSS-10431 - Given a task and the applicable work standards, describe the application of core work practices to perform the task IAW management's expectations.				
References Provided	none			LORT: Open	
Question Source (New, Modified, Bank)	New				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	Memory or Fundamental Knowledge	X	Comprehension or Analysis		
10CFR55 Content	55.41b		55.43b	3	
10CFR55 Explanation	Facility licensee procedures required to obtain authority for design and operating changes in the facility.				

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	14	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

96 **ID: 1248461** **Points: 1.00**

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

- Annunciator 10F-3-g, SVC WTR HI/TROUBLE, alarms.
- An Operator reports that the FAIL lamp on the Service Water Rad Monitor (RN01A) is LIT.
- Chemistry has been notified and determines that Service Water release rates are normal.

Which one of the following actions is required to allow Service Water operation to continue, in accordance with the Offsite Dose Calculation Manual (ODCM) and RAP-10F3g?

A. Verify the other Service Water Radiation Monitor is operable within 12 hours.

B. Collect and analyze Service Water effluent grab samples at least once per 24 hours.

C. Determine estimated service water pump flow rate at least once per 4 hours.

D. Collect and analyze two independent Service Water effluent grab samples and have two technically qualified individuals verify calculations and valving within 12 hours.

Answer: B

Answer Explanation

K&A	2.3.11 - Ability to control radiation releases. (4.3)		
Level: SRO	Tier: 3		Group:
General References	RAP 10F-3-g	ODCM	
Explanation	Proposed Answer: B Explanation: ODCM table 3.3.3.10-1 requires a Service Water effluent line radiation monitor to be operable. With the only installed Service Water effluent line radiation monitor inoperable, ACTION 112 applies and requires, "With no channels OPERABLE, effluent releases via this pathway may continue provided that, at least once per 24 hours, grab samples are collected and analyzed for radioactivity..." A. Plausible – Most process systems have built in redundancy. However, for Service Water there is only 1 effluent radiation monitor. If the applicant believes there is a backup radiation monitor, this choice is plausible. C. Plausible – This is patterned after ODCM ACTION 115. When no channel is operable, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Since there is no release in progress, this action does not apply. D. Plausible – This is patterned after ODCM ACTION 110. When no channel is operable, a sample is required to be taken and independently verified "BEFORE initiating a release", which is not the case here.		

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Lesson Plan Learning Objective/	2621. 828.0.0044 - SERVICE WATER SYSTEM SWS-00888 - Using the procedures, identify and interpret normal and abnormal operations of the Service Water System.				
References Provided	none			LORT: Open	
Question Source (New, Modified, Bank)	New				
Previous 2 NRC Exams (ILT Only)	No				
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X	
10CFR55 Content	55.41b		55.43b	4	
10CFR55 Explanation	Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.				
Justification for LORT questions with K/A values < 3.0	N/A				
Time to Complete:	1-2 minutes				
Point Value:	1				
System ID No.:	N/A	PRA:	No		
Safety Function(s):	9	☒ ILT			
Category(s) (LORT Only):	N/A	☐ LORT			

97 **ID: 1248462** **Points: 1.00**

The plant was at rated power when a **LOCA and ATWS** occurred. Plant conditions include the following:

- Reactor power is 15% and steady
- RPV water level indicates -16" and lowering
- Emergency Depressurization has been performed
- SP-17, Terminate and Prevent Injection, has been completed
- RPV Pressure has just lowered below the Minimum Steam Cooling Pressure (MSCP)

IAW the RPV Control - with ATWS EOP, which of the following systems shall the SRO direct **FIRST** to restore RPV water level **AND**, IAW the EOP User's Guide, which is the correct basis for this action?

A. Feed and Condensate IAW SP-19, Feedwater/Condensate and CRD System Operation, since it injects outside the core shroud.

B. Fire Water via the Core Spray System IAW SP-20, Low Pressure Injection During an ATWS, due to its ability to be throttled and controlled.

C. Core Spray System IAW SP-20, Low Pressure Injection During an ATWS, due to its ability to restore RPV water level faster than other injection systems.

D. Condensate Transfer via the Core Spray System IAW SP-20, Low Pressure Injection During an ATWS, due to its ability to throttle and is at a higher water purity than Fire Water.

Answer: A

Answer Explanation			
K&A	2.4.22 - Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations. (4.4)		
Level: SRO	Tier: 3		Group:
General References	RPV control - with ATWS	EOP User's Guide	

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Explanation	Proposed Answer: A Explanation: The question stem describes a condition where there is both a LOCA and ATWS. When ED is performed during an ATWS, pressure is allowed to lower below the MSCP, then makeup to the RPV commences via a series of preferred Safety Systems. Feed/Condensate and CRD are the FIRST priority since they inject outside the Core Shroud, allowing the cold water injected to warm and mix with borated water before entering the core. The first makeup source the SRO shall direct is Feed and Condensate. B. Plausible – Fire Water and Condensate Transfer via Core Spray are one of the next sources of water in line for makeup due to their ability to be throttled. Feed and Condensate has a higher priority though due to it injecting outside the core shroud where Fire Water and Condensate Transfer would inject cold water directly on top of the core. C. Plausible – The Core Spray system is the last source of makeup during an ATWS due to its injection of large quantities of cold unborated water injecting directly on the core. D. Plausible – Fire Water and Condensate Transfer via Core Spray are one of the next sources of water in line for makeup due to their ability to be throttled. Feed and Condensate has a higher priority though due to it injecting outside the core shroud where Fire Water and Condensate Transfer would inject cold water directly on top of the core.			
Lesson Plan Learning Objective/	2621.845.0.01B - RPV CONTROL-WITH ATWS EWA-03055 - Given a copy of RPV Control, describe in detail each step or conditional statement, including technical basis, and how to perform each step as required.			
References Provided	none			LORT: Open
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge		Comprehension or Analysis	X
10CFR55 Content	55.41b		55.43b	5
10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.			
Justification for LORT questions with K/A values < 3.0	N/A			

EXAMINATION ANSWER KEY

2016 SRO NRC Test

Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	10	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

98 **ID: 1248463** **Points: 1.00**

The plant was at rated power when a scram and an RPV isolation occurred. Present plant conditions are as follows:

- Isolation Condenser A was being used for cooldown
- Both Isolation Condensers are currently in Standby
- ISOL CONDENSER A LEVEL indicates 7.2' and steady
- ISOL CONDENSER B LEVEL indicates 7.4' and steady
- ISOL COND A SHELL indicates 208 °F and lowering
- ISOL COND B SHELL indicates 89 °F and steady

The following annunciators then alarmed 2 minutes later:

- ISOL COND – COND AREA TEMP HI
- RADIATION MONITORS – AREA MON HI (ARM C3, ISOLATION COND AREA indicates 9 mr/hr)
- RB ΔP LO

The Operator reports:

- **NO CHANGE** in the Isolation Condenser shell level indications.
- RB ΔP is -.14"w.g. and degrading .05"w.g. per minute

Which of the following states the potential impact in the next 2 minutes from operators report and the required SRO direction?

	Impact	SRO Direction
A.	Increase in dose to workers in the RB	Isolate **BOTH** ICs IAW Secondary Containment Control EOP
B.	Increase in dose to workers in the RB	Isolate IC-A **ONLY** IAW Secondary Containment Control EOP
C.	Increase in offsite radioactivity release	Isolate **BOTH** ICs IAW Radioactivity Release Control EOP
D.	Increase in offsite radioactivity release	Isolate IC-A **ONLY** IAW Radioactivity Release Control EOP

Answer: A

Answer Explanation

K&A	2.3.14 - Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (3.8)		
Level: SRO	Tier: 3		Group:
General References	**SCC EOP**		

<table>
<tr><td>Explanation</td><td colspan="4">Correct Answer A

Explaination A is correct and B is incorrect the plant was at rated power when a scram and RPV isolation occurred. With the MSIVs closed, Isolation Condenser is being used for cooldown (place in service, then remove, then place in service as required). The initial conditions given show that IC A was in service and is now back in standby. The provided annunciators, combined with no changes to the initial trends, show a steam leak into the RB in the vicinity of the Isolation Condensers in the RB: high area temperature (at max normal temperature of 160 °F), high radiation in the vicinity of the isolation condensers, and low RB ÄP. Entry into the Secondary Containment Control EOP is required. With a steam leak in the RB, dose to workers in the RB may rise. IAW the EOP, the leak should be isolated. Because the indications do not point to one condenser or the other as the leak source, both condensers should be isolated.

C and D are Incorrect. The given indications are not indicative of a tube leak in an isolation condenser (shell water level rising, shell water temperature rising). The ARM, by itself, could be indicative of a tube leak. These indications (rising shell water level, rising shell water temperature, ARM) would require entry into the Radiological release EOP, which in this case is not required. A tube leak in a condenser could lead to an increase in offsite release. Although the actions are correct, the procedure guidance is not correct.</td></tr>
<tr><td>Lesson Plan

Learning Objective/</td><td colspan="4">2621.845.0.11, Secondary Containment Control

SCC-3082, Using the Secondary Containment Control EOP, evaluate the technical basis for each step and apply this evaluation to determine the correct course of action under emergency conditions.</td></tr>
<tr><td>References Provided</td><td>none</td><td></td><td colspan="2">LORT: Open</td></tr>
<tr><td>Question Source (New, Modified, Bank)</td><td colspan="4">Bank</td></tr>
<tr><td>Previous 2 NRC Exams (ILT Only)</td><td colspan="4">No</td></tr>
<tr><td>Cognitive Level</td><td>Memory or Fundamental Knowledge</td><td></td><td>Comprehension or Analysis</td><td>X</td></tr>
<tr><td>10CFR55 Content</td><td>55.41b</td><td></td><td>55.43b</td><td>4</td></tr>
<tr><td>10CFR55 Explanation</td><td colspan="4">Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.</td></tr>
</table>

Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	9	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

99 **ID: 1248464** **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.

Which one of the following is correct regarding the scheduled removal of CRD Pump NC08A from service, IAW WC-OC-101-1001, On-Line Risk Management and Assessment?

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 **cannot** be removed from service as scheduled.

Answer: D

Answer Explanation			
K&A	2.2.17 - Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator. (3.8)		
Level: SRO	Tier: 3		Group:
General References	WC-OC-101-1001		
Explanation	Proposed Answer: D 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. A. Plausible – This is an activity associated with plant status risk colors when the risk is higher than Green (green being the lowest risk). This action is required for yellow risk. B. Plausible – This is an activity associated with plant status risk colors when the risk is higher than Green (green being the lowest risk). This action is required for orange risk. C. Plausible – This is an activity associated with plant status risk colors when the risk is higher than Green (green being the lowest risk). This action is required for yellow risk.		
Lesson Plan Learning Objective/	2612.DBIG.0011 - On-Line Risk and Shutdown Safety Management Program 2612.DBIG.0011-5 - Describe the purpose of the On-Line Risk Management and Assessment Program.		

EXAMINATION ANSWER KEY

2016 SRO NRC Test

References Provided	none			LORT: Open
Question Source (New, Modified, Bank)	Bank			
Previous 2 NRC Exams (ILT Only)	No			
Cognitive Level	Memory or Fundamental Knowledge	X	Comprehension or Analysis	
10CFR55 Content	55.41b		55.43b	5
10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.			
Justification for LORT questions with K/A values < 3.0	N/A			
Time to Complete:	1-2 minutes			
Point Value:	1			
System ID No.:	N/A	PRA:	No	
Safety Function(s):	14	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		

100 **ID: 1248465** **Points: 1.00**

The plant was at rated power when an event occurred. Present plant conditions are as follows:

- COND B FLOW HI POSSIBLE RUPTURE annunciator is in alarm
- SHELL B LVL HI/LO annunciator is in alarm
- ISOL CONDENSER B LEVEL indicates 9.5'
- Isolation Condenser B **CANNOT** be isolated
- RADIATION MONITORS – OFFGAS HI annunciator is in alarm
- RADIATION MONITORS – OFFGAS HI-HI annunciator is in alarm
- MAIN STEAM – RAD HI annunciator is in alarm
- MAIN STEAM LINE RAD MONITORS indicate > 3000 mr/hr
- RPS 1 and RPS 2 SCRAM SOLENOIDS lights are de-energized
- Several Area Radiation Monitors in the Reactor Building are reading slightly above their high setpoint
- The Site Emergency Director has declared a General Emergency - EAL (RG1, Radiological Effluent)

Which of the following actions is **REQUIRED** and what is the associated basis for the action?

A. Emergency Depressurize the RPV IAW the Radioactivity Release Control EOP in order to protect secondary containment integrity.

B. Emergency Depressurize the RPV IAW the Radioactivity Release Control EOP in order to reduce the release rate outside of the containments.

C. Depressurize the RPV to maintain the cooldown rate below 100 °F/hr IAW the RPV Control – No ATWS EOP, in order to reduce the driving head of the leak.

D. Depressurize the RPV to maintain the cooldown rate below 100 °F/hr IAW the RPV Control – No ATWS EOP, in order to avoid exceeding two maximum safe values in the secondary containment

Answer: B

Answer Explanation			
K&A	2.4.23 - Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations. (4.4)		
Level: SRO	Tier: 3		Group:
General References	EOP User's Guide		

Explanation	Proposed Answer: B Explanation: The requirements to ED in the Rad Release EOP are: indications of fuel damage, and a General Emergency declared due to offsite dose (which has been declared and provided). Therefore, ED is required IAW the Rad Release EOP. The EOP User's guide describes the basis for this action as minimizing the release rate and placing the RPV and attached primary systems in the lowest possible energy state to reduce the driving head and flow of any primary systems that are discharging outside the containments. A. Plausible – This would be a correct basis for an ED if two max safe values were exceeded. The indications provided show a general rise in radiation levels in the reactor Building but not to the extent of max safe. C. Plausible – No ATWS EOP does direct establishing a normal cooldown, but this is overridden by the need to perform an ED from other EOPs. D. Plausible – No ATWS EOP does direct establishing a normal cooldown, but this is overridden by the need to perform an ED from other EOPs. Also, the basis listed corresponds to the Secondary Containment Control EOP.					
Lesson Plan Learning Objective/	2621. 845.0.12 - Radioactivity Release Control LP RRC-02483 - Using procedure Radioactivity Release Control, evaluate the technical basis for each step and apply this evaluation to determine the correct course of action under emergency conditions.					
References Provided	none				LORT: Open	
Question Source (New, Modified, Bank)	Modified (from 907023)					
Previous 2 NRC Exams (ILT Only)	No					
Cognitive Level	Memory or Fundamental Knowledge			Comprehension or Analysis	X	
10CFR55 Content	55.41b			55.43b	5	
10CFR55 Explanation	Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.					
Justification for LORT questions with K/A values < 3.0	N/A					
Time to Complete:	1-2 minutes					
Point Value:	1					
System ID No.:	N/A	PRA:	No			

Safety Function(s):	10	☒ ILT		
Category(s) (LORT Only):	N/A	☐ LORT		