Given:
Unit 2 is at 100% power with all systems aligned normally for this power level.

A pressurizer reference leg leak occurs that is causing:

- ALL Pressurizer pressure instruments to sense a CHANGE in pressure of 100 psi/min.
- 2LT459 to sense a CHANGE in level of 8%/min.

With these conditions, an Automatic Reactor Trip will occur in:

A. 1 minute and 30 seconds due to Pressurizer Pressure High

B. 3 minutes and 30 seconds due to Pressurizer Pressure Low

C. 4.0 minutes due to Pressurizer Level High

D. 4 minutes and 4 seconds due to Pressurizer Pressure Low.

Answer: B

Answer Explanation:

Meets KA. Requires examinee knowledge of Reactor Trip setpoints and coincidences to make up a reactor trip which will be indicated on the reactor trip status panel. In addition the examinee must distinguish how a leak will affect level and pressure channels associated with monitoring pressurizer values.

NOP is 2235psig. A reference line leak will cause indicated pressure to lower. Reactor trip on Lo Pzr Pressure setpoint is 1885. Difference is 350 psi. At 100 psi per minute change it will take 3.5 minutes.

Pressurizer pressure high is incorrect but is plausible if the examinee mistakenly thinks that a reference line break will cause an increase in pressure such as used in a d/p cell. The 1.5 minutes is the correct time to reach the High Pressurizer Pressure Reactor trip setpoint of 2385 if pressure were changing by 100 psi per minute.

This event would cause Pressurizer level high making this choice plausible and the stated time limit is correct as calculated from 60 to 92% but it will occur on only one channel making this incorrect.

Pressure Pressure Low, is correct but 4 minutes and 4 seconds would correspond to an RCS Pressure of 1829 psig which is the Safety Injection setpoint making this a plausible distractor.

ref. BAR 2-12-A1,B1, A2, C2 and BAR 2-11-A3

Question 1 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28249
User-Defined ID:	2012 NRC EXAM RO Q1
Cross Reference Number:	S.RP3-04
Topic:	2012 NRC Exam RO Question 1
Num Field 1:	RO 3.5
Num Field 2:	SRO 3.6
Text Field:	007K2.03
Comments:	**New question, Hi Cog., RO Level** **EPE: 007 Reactor Trip** **EK2 Knowledge of the interrelations between a reactor trip and the following:** (CFR 41.7 / 45.7) EK2.03 Reactor trip status panel 3.5 3.6

Associated objective(s):

EXPLAIN the relationship of RPS, RCS overpressure protection, Main Steam System overpressure protection and AMS in mitigating ATWS condition overpressure transients

Given:

- You are the Unit 1 Reactor Operator.
- Unit 1 is at 95% power with all control systems in automatic.
- A small break LOCA occurs resulting in a Reactor Trip and Safety Injection.
- You have transitioned from 1BEP-0, REACTOR TRIP OR SAFETY INJECTION UNIT 1, to 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT UNIT 1, AND have subsequently transitioned to 1BEP ES 1.1 S.I. TERMINATION UNIT 1.
- Normal charging flow has been established.
- The RH and SI pumps have been stopped.
- Containment pressure PEAKED at 4.1 psig and is LOWERING.

Step 11 of 1BEP ES-1.1 states "VERIFY ECCS FLOW NOT REQUIRED".

For which of the following conditions would you recommend starting additional ECCS pumps and returning to 1BEP-1, Loss of Reactor or Secondary Coolant?

A. ALL SG NR levels <10%

B. TOTAL FW flow to ALL SGs is <500 gpm

C. Pressurizer level is 20%

D. SPDS subcooling iconics displays "0"

Answer: D

Answer Explanation:

Meets KA. Requires examinee knowledge of starting additonal ECCS pumps in relationship to a small break LOCA

Per Step 11 of 1BEP ES-1.1, SI Termination Unit 1, and the associated Operator Action Summary page. If either RCS Subcooling is unacceptable OR Pressurizer Level is not greater than 12% (28% Adverse), manually start ECCS pumps and go to 1BEP-1 Loss of Reactor or Secondary Coolant. Per the background document, ensuring both Pzr level and subcooling requirements are met, ensures the RCS conditions are under the control of the operator.

The SG parameters are plausible based on EP-1 steps to "check if ECCS flow should be reduced" which contains SG level and total FW flow requirements

Ref: 1BEP ES-1.1 step 11 and WOG Background document pg. 16.

Question 2 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28250
User-Defined ID:	2012 NRC EXAM RO Q2
Cross Reference Number:	T.EP02-01-D
Topic:	2012 NRC Exam RO Q2
Num Field 1:	RO 4.1
Num Field 2:	SRO 4.3
Text Field:	009K3.04
Comments:	New Question W. Hochstetter 7/21/11 RO Level Low Cog **009 Small Break LOCA** **EK3 Knowledge of the reasons for the following responses as the apply to the small break LOCA**: (CFR 41.5 / 41.10 / 45.6 / 45.13) EK3.04 Starting additional charging pumps 4.1 4.3

Associated objective(s):

Without the use of EP-1, EP ES-1.1, 1.2, 1.3, 1.4: EXPLAIN the intent of and basis for each step in the procedure

Given:

- All Unit 2 Reactor Coolant Pump Seal Outlet Temperatures are 175 degrees F.
- A LOSS of seal cooling occurs to all RCPs.

How much further can Seal Outlet Temperatures be allowed to RISE before the affected RCPs must be stopped?

A. 9 degrees F.

B. 20 degrees F.

C. 50 degrees F.

D. 60 degrees F.

Answer: D

Answer Explanation:

Meets KA as the question requires the examinee to know required RCP Trip criteria on loss of seal cooling

235°F is called out as trip criteria for seal outlet temperature in1BOA RCP-2 Loss of Seal Cooling.

184°F is credible as it is listed in BOP RC-1 as criteria to open the seal bypass valve.
195°F is credible as it is listed in BOP RC-1 as criteria to trip the RCP on high motor bearing temperature.
225°F is credible as it is listed on 1BOA RCP-2 as RCP trip criteria associated with lower radial bearing temperature.

Ref: BOA RCP-2 step 1, BOP RC-1 pg 4 and 8

Question 3 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28335
User-Defined ID:	2012 NRC EXAM RO Q3
Cross Reference Number:	T.OA27-07
Topic:	2012 NRC Exam RO Question #3
Num Field 1:	RO 3.7
Num Field 2:	SRO 3.7
Text Field:	015/017AA2.10
Comments:	New Question, RO Level, High Cog. **015/017 Reactor Coolant Pump (RCP) Malfunctions** **AA2. Ability to determine and interpret the following as they apply to** **the Reactor Coolant Pump Malfunctions (Loss of RC Flow): (CFR 43.5 / 45.13)** AA2.10 When to secure RCPs on loss of cooling or seal injection . . . . 3.7 /3.7

Associated objective(s):

STATE the temperature limits associated with proper reactor coolant pump operation

Assume the following on Unit 1:

- 100% power normal operation and 120 gpm Letdown.
- VCT Level is 50%
- Unit has maximum allowable identified leakage per Tech. Specs.
- VCT level channel 112 fails HIGH
- NO Operator Action is taken

What is the status of charging flow, pressurizer level and RCS boron concentration 1 hour after the failure?

	Charging Flow	Pressurizer Level	RCS Boron Concentration
A.	Lower	Same	Higher
B.	Same	Lower	Higher
C.	Same	Same	Same
D.	Lower	Lower	Same

Answer: D

Answer Explanation:

Meets KA as the question requires the examinee to access charging flow and pressurizer level changes based on a loss of reactor coolant makeup

LT-112 failing high prevents auto make-up from occurring to the VCT. The 2 out of 2 signal (LT112 and LT185) @ 5% VCT level that is required for the swap over of the charging pump suction to the RWST will not function. LT112 failed high will also cause letdown to divert to the HUT. VCT level at 50% is approximately 1000 gallons of make-up available. VCT level will drain to 0% in about 8.3 mins. due to no letdown and 132 gpm charging. Note 12 gpm will be added to the VCT by seal leak-off. After approximately 8 minutes (VCT level is about 0 and the running CV pump would cavitate causing a flow reduction into the RCS) Pressurizer level will start to drop from 60% to 17% (due to continued letdown and no charging) at about 1% per minute. Total time now is about 51 minutes. At this point with essentially no charging and no letdown (except seal return flow) and a 10 gpm leak Pzr level will continue to drop about 5% over the remaining hour.

All distracters are plausible based on misconceptions that (1) a swapover would occur which would have lower Tave causing programmed pressurizer level to lower and hense lowering charging flow. (2) a swapover would occur and enough boron was placed into the RCS to lower RCS Tave which would cause a reduction in pressurizer level based on contraction of the RCS without affecting charging flow and (3) that VCT make-up would not be affected because the controlling feature comes off of LT-185 or it takes both channels to affect the swap.

Ref: BAR 1-9-A2

Question 4 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28251
User-Defined ID:	2012 NRC EXAM RO Q4
Cross Reference Number:	S.CV2-11
Topic:	2012 NRC Exam RO Question 4
Num Field 1:	RO 3.0
Num Field 2:	SRO 3.4
Text Field:	022AK1.03
Comments:	Bank Question RO Level High Cog **APE: 022 Loss of Reactor Coolant Makeup** IMPORTANCE **K/A NO. KNOWLEDGE** RO SRO **AK1. Knowledge of the operational implications of the following concepts as** **they apply to Loss of Reactor Coolant Makeup** AK1.03 Relationship between charging flow and PZR level 3.0 3.4

Associated objective(s):

Given a Reactor Makeup Control System operating mode and various plant conditions, PREDICT how plant parameters will respond to related component or controller failures

Given:

- Unit 2 is in MODE 5.
- RCS pressure is 100 psig.
- Operating CC pump discharge pressure is 115 psig.
- 2A RH pump is running in the shutdown cooling mode.

A large tube leak develops in the 2A RH Heat Exchanger.

How and why is the Unit 2 Component Cooling Water system affected?

CC pump discharge pressure...

A. Rises due to RISING CC surge tank level.

B. Drops due to LOWERING CC surge tank level.

C. Drops due to pumping a GREATER flowrate.

D. Rises due to LOWER CC system back pressure.

Answer: A

Answer Explanation:

Meets K/A. Requires knowledge of effect of heat exchanger tube rupture, knowledge of RH pump effects on system pressure and the interrelationship of RH and CC in the RH Heat exchanger.

With RH pump running, RH pump discharge pressure will be in excess of 200 psig because RCS pressure of 100 psig is added to pump d/p to obtain RH pressure in the heat exchanger, which is the location of the leak. RH (RCS) will leak into the CC system causing an increase in surge tank level and hence a higher CC pump discharge pressure.

The 2 distractors stating that CC pump discharge pressure lowers are considered plausible because as stated in the stem, CC system pressure is higher than RCS pressure which may cause the examinee to think that the leak will result in CC leaking out of the CC system.

CC system pressure rising based on lower CC system backpressure is plausible if the examinee doesn't understand a centrifugal pump characteristic curve.

Ref: BAR 2-2-A5 which also refers the operator to BOA PRI-10, Loss of RH Cooling, and to BOA PRI-6, Component Cooling Water Malfunction. High Surge tank level is a possible entry condition to Loss of RH which then directs the operator to continue in PRI-10 and and Go to BOA PRI-6.

Question 5 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28279
User-Defined ID:	2012 NRC EXAM RO Q#5
Cross Reference Number:	S.RH1-07
Topic:	2012 NRC Exam RO Question #5
Num Field 1:	RO 2.7
Num Field 2:	SRO 2.7
Text Field:	025AK2.03
Comments:	New Question High Cog RO Level **025 Loss of Residual Heat Removal System (RHRS)** **AK2. Knowledge of the interrelations between the Loss of Residual Heat** **Removal System and the following:** **(CFR 41.7 / 45.7)** AK2.03 Service water or closed cooling water pumps 2.7 2.7

Associated objective(s):

DESCRIBE the available Control Room Instrumentation for monitoring temperature, pressure, and flow in the RH System

Given:
- Unit 1 at 100% power.
- All systems are normally aligned.
- Pressurizer Pressure Control Channel Select Switch is in 455/456 position.
- Annunciator 1-12-A1, PZR PRESS LOW RX TRIP STPT ALERT, is LIT.
- Annunciator 1-12-C1, PZR PRESS CONT DEV LOW HTRS ON, is LIT.
- 1PT455 has failed LOW.

Based on the above conditions, and with NO operator actions, which of the following will be observed on the control boards?

A. PZR PORV 1RY456 is cycling.

B. PZR PORV 1RY455A is cycling.

C. Both spray valves, 1RY455B & C, are FULL OPEN.

D. P-11 is LIT on the Bypass Permissive Panel.

Answer: A

Answer Explanation:

Question meets KA - question requires examinee to determine the response of PZR PORV's, heaters and spray to a pressurizer control malfunction.
Normal PZR pressure control alignment is channel 455/456. If the controlling channel (455) failed low, PZR heaters would energize and RCS pressure would rise. PZR spray valves and PORV 455A would not open. PZR PORV 456 would open and cycle to maintain RCS pressure. Annunciator 1-12-A1 and 1-12-C1 would come in due to 455 failed low.

The other choices are all plausible based on the examinees misconception of the failure and interface with the control system.

Ref: Annunciator 1-12-A1 and 1-12-C1 and Pressurizer L-P page 47.

Question 6 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28336
User-Defined ID:	2012 NRC EXAM RO Q6
Cross Reference Number:	3D.OA-11-A
Topic:	2012 NRC Exam RO Question 6
Num Field 1:	RO 4.0
Num Field 2:	SRO 3.9
Text Field:	027AA1.01
Comments:	Bank Question, HI Cog, RO Level **027 Pressurizer Pressure Control System (PZR PCS) Malfunction** **AA1. Ability to operate and / or monitor the following as they apply to** **the Pressurizer Pressure Control Malfunctions:** **(CFR 41.7 / 45.5 / 45.6)** AA1.01 PZR heaters, sprays, and PORVs 4.0 3.9

Associated objective(s):

Given a set of plant conditions or parameters indicating a malfunction in the Pressurizer Pressure Control System, DISCUSS the integrated plant response to the event/casualty with no operator action

Given:

- The reactor was initially at 100% power.
- All systems are configured for a full power line up.
- A transient has occurred that caused reactor trip setpoints to be exceeded.
- An automatic reactor trip does NOT occur.
- Your attempts to trip the reactor manually from the control room fail.

Which choice below will result in a Reactor TRIP? (consider each choice SEPARATELY)

A. Place MG set B generator side breaker in "pull to lock".

B. Open MG Set B motor side breaker by depressing the switchgear manual TRIP pushbutton.

C. Locally trip RTB by depressing the switchgear manual TRIP pushbutton.

D. Remove the DC Control power fuse for RTB.

Answer: C

Answer Explanation:

Meets K/A. RO Level. Requires knowledge of local reactor trip actions and Rod Drive system design. Question steam places the operator in an ATWS condition and requires knowledge of how to trip the reactor locally using switchgear trip pushbuttons.

Distractors involving opening of 1 MG set motor or generator breaker will not result in a trip because 2 MG sets are normally operating, however this is plausible based on ATWS actions. Removing DC control power fuses will not result in the Reactor Trip breakers opening however, it is considered plausible based on an under voltage trip (from SSPS) for the reactor trip breakers.

The correct answer is correct based on only 1 reactor trip breaker must open to cause a reactor trip.

Ref. 6E-1-4030RD07, lower right hand portion "key diagram".

Question 7 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28275
User-Defined ID:	2012 NRC EXAM RO Q7
Cross Reference Number:	T.FR01-07
Topic:	2012 NRC Exam RO question #7
Num Field 1:	RO 4.5
Num Field 2:	SRO 4.5
Text Field:	029EA1.08
Comments:	New Question High Cog. RO level **029 Anticipated Transient Without Scram (ATWS)** **EA1 Ability to operate and monitor the following as they apply to a ATWS:** EA1.08 Reactor trip switch pushbutton . . . . 4.5 4.5 (CFR 41.7 / 45.5 / 45.6)

Associated objective(s):

ANALYZE a given set of conditions and DETERMINE the appropriate operator actions to respond to an ATWS event

Given:

- Unit 1 was operating at HFP (100% power) with all systems aligned normally.
- The 10 highest Core Exit thermocouple (CETC) temperature readings were 617 degress F.
- 12 minutes ago a Reactor Trip and Safety Injection were initiated due to a Steam Generator Tube Rupture.
- The crew has exited 1BEP-0, REACTOR TRIP OR SAFETY INJECTION UNIT 1 and has transistioned to 1BEP-3, STEAM GENERATOR TUBE RUPTURE, UNIT 1.

Currently:
- RCS Pressure is 1785 psig
- The 10 highest Core Exit thermocouple readings are 548 degrees F.
- The lowest ruptured S/G pressure (prior to cooldown) is 1085 psig

The crew is at step 6 preparing to Initiate RCS Cooldown to 516 degrees F by dumping steam at the Maximum Rate with the intent of eventually lowering RCS Pressure to current S/G pressure.

Of the following, Unit 1 was at MINIMUM RCS Subcooling ...

A. While at HFP.

B. At HZP temperature and SI setpoint for RCS pressure.

C. Immediately PRIOR to the initial cooldown.

D. At the completion of the FIRST depressurization.

Answer: A

Answer Explanation:

Meets K/A. Question places the examinee in a SGTR and requires the use of steam tables to calculate RCS subcooling.

Prior to the event Th=617, Pzr = 653 Subcooling = 36 degrees

Immediately prior to initial cooldown is not correct because 1800 psia has a sat. temp of 621 degress, RCS is currently 548 degress leaving 73 degrees of subcooling.

After the initial depressurization is not correct because RCS pressure of 1100 psia corresponds to 556 degrees, RCS temp is 516 degrees resulting in 40 degrees subcooling.

At the time of trip and SI (assuming 557 degrees RCS temp) is not correct. 1829 psig corresponds with approximately 623 degrees. 623-557 = 66 degress of subcooling.

Ref: Steam tables, saturated portion.

Question 8 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28273
User-Defined ID:	2012 NRC EXAM RO Q8
Cross Reference Number:	T.EP04-08
Topic:	2012 NRC Exam RO Question #8
Num Field 1:	RO 3.1
Num Field 2:	SRO 3.4
Text Field:	038K1.01
Comments:	New Question High Cog RO Level **038 Steam Generator Tube Rupture (SGTR)** K/A NO. KNOWLEDGE RO SRO **EK1 Knowledge of the operational implications of the following concepts as they apply to the SGTR:** (CFR 41.8 / 41.10 / 45.3) EK1.01 Use of steam tables 3.1 3.4

Associated objective(s):

ANALYZE a set of plant conditions, and DETERMINE the required operator actions in responding to a SGTR, per _EP-3, _EP ES-3.1, 3.2 or 3.3

Given:
- Unit 2 was at 100% power.
- All systems were normally aligned.
- An inadvertent FW Isolation occurred.
- The crew has implemented 2BEP-0, REACTOR TRIP OR SAFETY INJECTION, and has transitioned to 2BEP ES-0.1, REACTOR TRIP RESPONSE.

While performing 2BEP ES-0.1, step 1, Check RCS Temperatures, the following indications are noted:
- ALL RCPs are running.
- RCS Tave is 553°F and LOWERING.
- ALL SG NR levels are 5% and RISING.
- ALL Steam Dumps are CLOSED.
- ALL SG Blowdown Isolation valves are CLOSED.
- BOTH AF Pumps are running and delivering 1200 GPM TOTAL flow to the SGs.
- Containment Pressure is 0.4 psig and STABLE.

Based on the above indications, the FIRST action the crew will take is:

A. close ALL MSIVs and MSIV bypass valves.

B. transition to 2BFR-H.1, RESPONSE TO LOSS OF HEAT SINK.

C. initiate emergency boration of the RCS per 2BOA Pri-2

D. throttle AF flow control valves while maintaining at least 500 GPM total flow to the SGs.

Answer: D

Answer Explanation:

Question meets KA - question requires examinee determine AF adjustments needed to maintain proper T-ave and S/G level during a Loss of Main Feedwater event.
With RCS temperature lowering less than 557°F, 2BEP ES-0.1 step 1 RNO directs the operators to stop dumping steam (already done w/steam dumps closed), isolate blowdown (done), then throttle AF flow to greater than 500 gpm. The step is a CAS, thus RO required knowledge.
Closing MSIVs action would be taken if other actions were ineffective. Question stem asks for first action.
The correct answer is correct see explanation above.
SG levels are below 2BFR-H.1 transition level, but AF flow is met making this incorrect.
emergency boration follows reducing AF Flow and is incorrect because the stem asks the first action

Ref: 2BEP ES-0.1 step 1 RNO

Question 9 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28258
User-Defined ID:	2012 NRC EXAM RO Q9
Cross Reference Number:	T.EP01-03
Topic:	2012 NRC Exam RO Question 9
Num Field 1:	RO 4.0
Num Field 2:	SRO 4.3
Text Field:	054AA2.06
Comments:	Bank Question 2011 Braidwood Cert Exam Hi Cog RO Level **054 Loss of Main Feedwater (MFW)** **AA2. Ability to determine and interpret the following as they apply to** **the Loss of Main Feedwater (MFW):** **(CFR: 43.5 / 45.13)** AA2.06 AFW adjustments needed to maintain proper T-ave. and S/G level 4.0 4.3

Associated objective(s):

Given the procedure, be able to DESCRIBE the Intent/Basis of each step and how it is performed

From the following, choose the ENTRY CONDITION and IMMEDIATE ACTIONS STEPS for 1BCA 0.0, LOSS OF ALL AC POWER, UNIT 1.

	Entry Condition	Immediate Actions
A.	Buses 141 AND 142 deenergized.	1) Verify negative SUR on IR NIs 2) Verify AF Flow > 500gpm
B.	BOTH SATs deenergized.	1) Verify Reactor Trip Breakers OPEN 2) Isolate Steam Lines
C.	Buses 141 AND 142 deenergized.	1) Verify Reactor Trip Breakers OPEN 2) Isolate Steam Lines
D.	BOTH SATs deenergized.	1) Verify negative SUR on IR NIs 2) Verify AF Flow > 500gpm

Answer: C

Answer Explanation:

Meets K/A. Examinee must know entry conditions and immediate actions of BCA 0.0.
The correct answer lists the entry conditions and both immediate action steps.
The entry conditions are all plausible because it contains the 4KV buses or their normal power supply .
The Immediate action steps are all plausible because they are procedure steps.

Ref: 1BCA 0.0 symptom or entry conditions section and Step 1 of the procedure.

Question 10 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28283
User-Defined ID:	2012 NRC EXAM RO Q10
Cross Reference Number:	T.CA1-01
Topic:	2012 NRC Exam RO Question #10
Num Field 1:	RO 4.6
Num Field 2:	SRO 4.8
Text Field:	055 G 2.4.1
Comments:	New Question Low Cog **055 Loss of Offsite and Onsite Power (Station Blackout)** **2.4 Emergency Procedures / Plan** 2.4.1 **Knowledge of EOP entry conditions and immediate action steps.** | (CFR: 41.10 / 43.5 / 45.13) | IMPORTANCE RO 4.6 SRO 4.8

Associated objective(s):

LIST all Immediate Actions for CA-0.0

Given:

- Unit 1 is at rated power, with all systems normally aligned.
- A loss of 120 VAC Instrument Bus 111 occurs.
- Shortly afterward, a valid SI signal is actuated.

With NO Operator action, which of the following describes the position and indication of 1CV112D, RWST to U-1 CHG PPS SUCT ISOL VLV one minute after the SI signal?

1CV112D is…

A. CLOSED, and indicates CLOSED.

B. OPEN, but both OPEN and CLOSED indications are DARK.

C. OPEN, and indicates OPEN

D. CLOSED, but both OPEN and CLOSED indications are DARK.

Answer: A

Answer Explanation:

Meets K/A. Requires examinee to assess the position of charging pump suction valve from RWST following a loss of vital instrument power.

The loss of the 120VAC Instrument bus results in the loss of the ‘A’ Train of Slave Relays in SSPS. This directly affects the 1CV112D, in that it will not receive an automatic signal to reposition to its SI position. Thus it will stay closed, until stroked by the operator, (which has been precluded by the ‘With NO Operator Action’ clause in the stem). The position indication lights are not affected, and will indicate that the valve is closed (closed light lit, and open light dark) because these lights are fed from 480V MCC. This valve is a MOV.

Open and indicates open is plausible if the examinee thinks this valve will be unaffected.
Closed, but both open and closed indications are dark is plausible if the examinee thinks that this valves is powered from Instr. Bus 111
Open and open and closed indication is dark is plausible if the examinee thinks the valve is controlled by an AC solenoid (fail open) fed from Inst. Bus 111.

Ref: 1BOA Elec-2 Table A 1.a which states that Train A ESF loads will not auto actuate or reset.

Question 11 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28254
User-Defined ID:	2012 NRC EXAM RO Q11
Cross Reference Number:	T.OA02-03
Topic:	2012 NRC Exam RO question 11
Num Field 1:	RO 3.3
Num Field 2:	SRO 3.5
Text Field:	057 AA2.07
Comments:	New Question High Cog RO Level **057 Loss of Vital AC Electrical Instrument Bus** **AA2. Ability to determine and interpret the following as they apply to** **the Loss of Vital AC Instrument Bus:** **(CFR: 43.5 / 45.13)** AA2.07 Valve indicator of charging pump suction valve from RWST 3.3 3.5

Associated objective(s):

ANALYZE a given set of plant conditions and DETERMINE the required actions per PER 1/2BOA ELEC-2, Loss of Instrument Bus

Given the following plant conditions on Unit 1:

- Reactor power is 50% and stable.
- Annunciator 1-21-E10 "125V DC DIST PNL 111/113 VOLT LOW" is LIT.
- DC bus 111 voltmeter = 0 volts.

Based on the above indications answer the following:

- Which DC power components have been affected?
- What is the status of the unit in 1 hour assuming NO operator action is taken?

	Affected DC components	Unit Status
A.	ONLY Bus 111 deenergized	on line
B.	111 Battery Charger deenergized	tripped
C.	Buses 111 AND 113 deenergized	tripped
D.	Buses 111 AND 113 deenergized	on line

Answer: C

Answer Explanation:

Meets K/A. Requires examinee to understand the DC Electrical system and recognize when the system has encountered a malfunction. In addition the examinee needs to know the affects of a loss of DC to continued power operation.

The combination of the annunciator with 0 Volts on the meter indicates a loss of DC bus 111. Meter indication >0 with the annunciator "lit" would indicate a loss of DC bus 113 only. The loss of the associated DC battery charger is considered a plausible distractor as it is the normal power supply to the DC Busses.

Ref: 1BOA Elec-1 step 1, the reactor will trip due to a loss of main feedwater to all unit 1 SGs.

Question 12 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28284
User-Defined ID:	2012 NRC EXAM RO Q12
Cross Reference Number:	3D.OA-23-A
Topic:	2012 NRC Exam RO Question 12
Num Field 1:	RO 4.1
Num Field 2:	SRO 4.3
Text Field:	058 G 2.4.45
Comments:	New Question, High Cog **058 Loss of DC Power** 2.4.45 **Ability to prioritize and interpret the significance of each annunciator or alarm.** | (CFR: 41.10 / 43.5 / 45.3 / 45.12) IMPORTANCE RO 4.1 SRO 4.3

Associated objective(s):

Given a set of plant conditions or parameters indicating a Loss of a DC Bus, DISCUSS the integrated plant response to the event/casualty with no operator action

Both Units are at 100% power, with the 1A and 2B SX pumps running.

The following annunciators alarm:

- 1-2-A1, SX PUMP TRIP
- 1-2-A2, SX PUMP DISCHARGE HEADER PRESSURE LOW

In accordance with 1BOA Pri-7, ESSENTIAL SERVICE WATER MALFUNCTION, the Unit 1 NSO should verify...

A. the 1B SX pump automatically starts.

B. 1SX005 and 2SX005, CC HX 0 INLT VLVs, are OPEN, then manually start the 1B SX pump.

C. 1SX033 and 1SX034, SX PP 1A and 1B XTIE VLVs, are OPEN.

D. the RCFC inlet and outlet valves, 1SX016B and 1SX027B, are OPEN, then manually start the 1B SX pump.

Answer: D

Answer Explanation:

Meets K/A . Requires examinee knowledge of BOA Pri-7 and actions to start a standby SX pump, should the running pump fail.

Auto start is plausible if examinee, thinks this feature works similair to CC pumps.
Manually starting the pump after opening 1and 2 SX005 is considered plausible since the BAR lists these actions if no pump is running on the Unit.
Cross-tie with the other train is considered plausible because the procedure does address this if NO SX pump is able to be started on the unit.

Ref: 1BOA PRI-7, Essentual Service Water Malfunction, Step 1. and BARs 1-2-A1, A2, and B1.

Question 13 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28285
User-Defined ID:	2012 NRC EXAM RO Q13
Cross Reference Number:	T.OA18-03
Topic:	2012 NRC Exam RO Question 13
Num Field 1:	RO 4.0
Num Field 2:	SRO 4.2
Text Field:	062G2.4.11
Comments:	Bank Question, Low Cog **062 Loss of Nuclear Service Water** 2.4.11 **Knowledge of abnormal condition procedures.** | (CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 4.0 SRO 4.2

Associated objective(s):

ANALYZE a given set of plant conditions and DETERMINE the required actions per 1/2BOA PRI-7/1BwOA PRI-8, Essential Service Water Malfunction

Given:

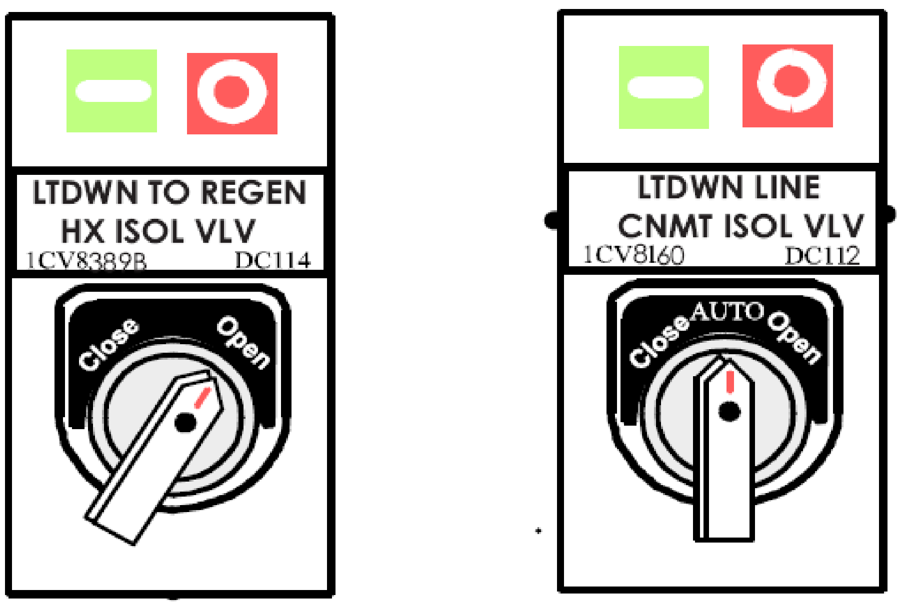
- Unit 1 is operating at 75% power
- The 1B Regen Heat Exchanger is in service.

The following sequence of events occurs:

- The control power fuse for 1IA065, INSTRUMENT AIR OUTSIDE CNMT ISOL VLV, blows in 1PM11J.
- Instrument air to Unit 1 containment is isolated and ALL affected CVCS valves in containment are in their FAIL position.
- 5 minutes later, the defective fuse is replaced and 1IA065 is re-opened.
- Instrument air is fully restored to the containment.
- Charging flow has been re-established to the 1B Regen Heat Exchanger.

The control switches depicted below have not changed state since the event occurred.

When instrument air was restored to the containment, what will be the positions of 1CV8389B and 1CV8160?



	1CV8389B is ...	1CV8160 is ...
A.	open	open
B.	closed	open
C.	closed	closed
D.	open	closed

Answer: D

Answer Explanation:

Meets KA. Requires examinee to predict CVCS valve position changes during restoration of IA to the containment.

The correct answer is correct because the 1CV8389B C/S is a maintain contact C/S. When air is restored no switch manipulation is needed to restore the valve to the original open position provided charging is reestablished to the heat exchanger first, as stated in the stem. 1CV 8160 is a spring return to center (auto) position. This circuit has a "seal-in" contact so the valve is maintained open, normally. When air pressure is lost, the valve fails closed and the seal in is lost. To reopen the valve the operator must take the switch to open until the closed light goes out and then release the control switch to re-establish the seal-in.
All distractors are plausible based on the students understanding of:
Valve failure positions on loss of air
understanding of control switch operations
understanding of seal in circuits
understanding of which valves are AOVs or MOVs

Ref: 6E-1-4030 CV28 and CV27

Question 14 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28341
User-Defined ID:	2012 NRC EXAM RO Q14
Cross Reference Number:	3C.IA-01-A
Topic:	2012 NRC Exam RO Question 14
Num Field 1:	RO 2.9
Num Field 2:	SRO 3.1
Text Field:	065AA1.03
Comments:	New Question, High Cog. RO Level **065 Loss of Instrument Air** **AA1. Ability to operate and / or monitor the following as they apply to** **the Loss of Instrument Air:** **(CFR 41.7 / 45.5 / 45.6)** AA1.03 Restoration of systems served by instrument air when pressure is regained 2.9 3.1 ref: 4030 CV27 (8389B) and CV-28 (8160)

Associated objective(s):

Given a set of plant conditions or parameters, DETERMINE whether the Instrument Air System is operating correctly.

Unit 1 has experienced a small break LOCA.

When performing 1BCA-1.2, LOCA OUTSIDE CONTAINMENT, why are RH components isolated BEFORE the other ECCS components?

A. To isolate the MOST likely location.

B. To ensure RCS injection flow is maintained during leak identification.

C. To isolate the lowest setpoint suction relief valves of all ECCS pumps.

D. To prevent damage to the RH system so it remains available for long term cooling.

Answer: A

Answer Explanation:

Meets KA. Requires examinee to know the reason why the RH system is first checked for leaks during a LOCA outside of containment.

Correct answer: LOCA outside containment is most likely to occur in RHR piping due to low system design pressure.
All distractors are plausible reasons as to why RH is isolated first, but are incorrect.
RH pp suction relief valves lift at 450 psig, while the same components for CV and SI pumps lift at 220 psig, making this distractor wrong but plausible.

Ref: WOG Background document CA-1.2 pg 6 of 10.

Question 15 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	28012
User-Defined ID:	2012 NRC EXAM RO 15
Cross Reference Number:	T.CA2-03
Topic:	Background for RH isolation
Num Field 1:	3.8
Num Field 2:	3.8
Text Field:	WE04EK3.3
Comments:	Bank Question,Low Cog, RO Level **E04 LOCA Outside Containment** **EK3. Knowledge of the reasons for the following responses as they apply to** **the (LOCA Outside Containment)** (CFR: 41.5 / 41.10, 45.6, 45.13) EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations. IMPORTANCE RO 3.8 SRO 3.8

Associated objective(s):

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Given the following plant conditions:

- Unit 1 was at 100% power.
- A large break RCS LOCA occurred.
- While performing 1BEP ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, 1SI8811A AND 1SI8811B, Containment Sump Isolation Valves, would NOT open in auto or manual.

The operating crew has implemented 1BCA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

Under these conditions, the reason 1BCA-1.1 directs establishing only one train of SI flow is to...

A. Allow initiating blended makeup flow to the suction of the charging pumps.

B. Reduce the RCS cooldown rate to less than 100°F/hr when dumping steam at maximum rate.

C. Reduce injection flowrate to what can be made up to the RWST by ALL available sources.

D. Reduce the RWST depletion rate to delay stopping all pumps taking suction from the RWST.

Answer: D

Answer Explanation:

Meets KA. Requires examinee to know the reason why only 1 train of ECCS flow is utilized while implementing Loss of Emergency Coolant Recirculation

Correct answer because a loss of ECR requires establishing one train of SI flow to minimize RWST depletion.

All distractors are considered plausible because they either limit the cooldown(preserve liquid inventory) rate or initiate make-up sources to the ECCS system.

Ref: WOG Background Doc. CA-1.1 Step 5 (pg.11), Step 11 (pg. 21) and step 12 (pg 22)

Question 16 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	28013
User-Defined ID:	2012 NRC EXAM RO 16
Cross Reference Number:	T.CA2-03
Topic:	BCA 1.1 RWST preservation
Num Field 1:	3.8
Num Field 2:	3.8
Text Field:	WE11EK3.3
Comments:	Bank Question, Low Cog., RO level WOG background document for CA-1.1 **E11 Loss of Emergency Coolant Recirculation** **EK3. Knowledge of the reasons for the following responses as they apply to** **the (Loss of Emergency Coolant Recirculation)** (CFR: 41.5 / 41.10, 45.6, 45.13) EK3.3 Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations. IMPORTANCE RO 3.8 SRO 3.8

Associated objective(s):

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Given:

- Unit 1 is experiencing a loss of heat sink condition with the following plant parameters:
  - ALL ECCS pumps are RUNNING.
  - ALL SG WR levels are 50%.
  - RCS pressure is 2200 psig.
  - Containment pressure is 1.4 psig.

- The crew is performing 1BFR-H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK at step 4, trying to re-establish AF flow.

With the above conditions, which one of the following conditions would require the crew to immediately initiate Bleed and Feed?
(consider each choice separately)

A. RCS pressure rises to 2300 psig.

B. ALL SG WR levels drop to 4%

C. Containment pressure rises to 5.2 psig.

D. 1A CV pump trips.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of conditions signals (MCR indications) that would require bleed and feed (operational implication)
1BFR-H.1 OAS page list bleed and feed criteria after performance of step 3 as any of the following.
WR SG level <27% (43% adverse) in any 3 SGs.
RCS pressure >2335 due to loss of heat sink.
No CV pumps available.

Choice D is incorrect, pressure still below 2335#.
Choice C is incorrect, containment would be adverse, but current SG levels at 50% would not meet bleed and feed criteria.
Choice B is correct, levels are below normal values of 27%.
Choice A is incorrect, 1B CV pp would still be running.

Ref: 1BFR-H.1 OAS page

Question 17 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28269
User-Defined ID:	2012 NRC EXAM RO Q17
Cross Reference Number:	T.MI08-03
Topic:	2012 NRC RO Exam Question 17
Num Field 1:	RO 3.9
Num Field 2:	SRO 4.1
Text Field:	W/E05K1.03
Comments:	From 2009 Braidwood NRC Exam High Cog RO Level **E05 Loss of Secondary Heat Sink** **EK1. Knowledge of the operational implications of the following concepts as** **they apply to the (Loss of Secondary Heat Sink)** (CFR: 41.8 / 41.10, 45.3) EK1.3 Annunciators and conditions indicating signals, and remedial actions associated with the (Loss of Secondary Heat Sink). IMPORTANCE RO 3.9 SRO 4.1

Associated objective(s):

APPRAISE each operator-initiated recovery technique in it's ability to restore the Heat Sink Critical Safety Function

Given:

- Both units are operating at full power and in normal full power alignments.
- DEH is configured with Impulse Pressure-IN and MW-OUT
- The station has entered 0BOA ELEC-1, DEGRADED SWYD VOLTAGE UNIT 0, due to actual degrading voltage conditions.

How will U-1 Turbine Generator Megawatt output and the voltage regulator respond as customer demand RISES and system frequency DROPS?

(Assume main turbine speed lowers by 4 rpm due to the load change.)

	Megawatt OUTPUT will	Voltage Regulator OUTPUT will
A.	rise	rise
B.	remain the same	rise
C.	remain the same	remain the same
D.	rise	remain the same

Answer: A

Answer Explanation:

Meets K/A. Requires knowledge of Turbine Generator load and voltage control during grid disturbances.

As customer demand increases, load on the system increases causing the system frequency and voltage to lower. The turbine speed reduction will result in calling for the governor valves to open. This will be partially off-set by P-imp limiting further open valve movement. Per BGP 100-3 if speed varies by more than 2 rpm, governor valve and hence reactor power will be affected. As system load increases voltage will lower. The voltage regulator, in auto, will increase output.

All distracters are plausible based on the examinee's knowledge of the interrelated systems and the affects that the grid places on unit operations.

Ref: BGP 100-3 Step E.2.s. (pg 15)

Question 18 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28286
User-Defined ID:	2012 NRC EXAM RO Q18
Cross Reference Number:	3D.OA-48-C
Topic:	2012 NRC exam RO Question 18
Num Field 1:	RO 3.6
Num Field 2:	SRO 3.7
Text Field:	077AK2.07
Comments:	New question, High Cog, RO level **077 Generator Voltage and Electric Grid Disturbances** **AK2. Knowledge of the interrelations between Generator Voltage |** **and Electric Grid Disturbances and the following: |** **(CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)** AK2.07 Turbine / generator control... 3.6 3.7

Associated objective(s):

Given a set of plant conditions or parameters indicating a 345 KV Grid or Voltage Regulator Instability condition and a set of plant procedures, IDENTIFY the correct procedure(s) to be utilized and DISCUSS required operator actions

While recovering a dropped Group 1 Rod on Control Bank C in accordance with 1BOA ROD-3, Dropped or Misaligned Rod, annunciator 1-10-C6, "ROD CONTROL URGENT FAILURE" alarm is EXPECTED from Unit 1 Rod Control Power Cabinet....

A. 1AC

B. 1BD

C. 2AC

D. 2BD

Answer: C

Answer Explanation:

Meets KA. Requires examinee to determine rod drive logic circuit alarms/actions associated with the recovery of a dropped control rod.

During the dropped rod recovery the lift coils are de-energized to all the rods in the affected group EXCEPT the affected rod, and all the rods in the opposite group of the affected control bank. This causes a Rod Control Urgent Failure alarm due to a regulation (logic) failure in the 2AC power cabinet since demand current does not equal actual current for any rod in that group. Choices with BD rod bank power cabinets are plausible because it is a common misconception for students to confuse power supplies and logic cabinets with control banks and groups.

Ref: BAR 1-10-C6 and 1BOA ROD-3

Question 19 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28253
User-Defined ID:	2012 NRC EXAM RO Q19
Cross Reference Number:	S.RD1-15-A
Topic:	2012 NRC Exam RO Question #19
Num Field 1:	RO 2.5
Num Field 2:	SRO 2.8
Text Field:	APE003 AK2.05
Comments:	Bank question RO level Low Cog 2011 Braidwood NRC exam **APE: 003 Dropped Control Rod** **AK2. Knowledge of the interrelations between the Dropped Control Rod and** **the following:** **(CFR 41.7 / 45.7)** AK2.05 Control rod drive power supplies and logic circuits 2.5 2.8

Associated objective(s):

DETERMINE the reasons for receiving a rod control urgent failure alarm, for the following condition: Dropped Rod Retrieval Procedure

Given:

- The plant is at 100% power.
- All control systems are in a normal/automatic line-up.
- The controlling PZR level transmitter, LT 459 sticks at the full power value.

Which of the following describes the subsequent PZR level control/plant response, 6 hours after power is REDUCED to 80%. (Assume NO operator action is taken aside from initiating the ramp)

	Charging Flow	Letdown Isolates	Rx Trip on PZR Hi Level
A.	constant	no	no
B.	lowers	yes	yes
C.	lowers	no	yes
D.	lowers	no	no

Answer: B

Answer Explanation:

Meets K/A, requires knowledge of PZR level control and inputs to RPS.
When controlling channel fails at 60% level and power is ramped back to a lower level the PZR level program calculates a new lower setpoint. Since level is above setpoint charging flow lowers to minimum, resulting in lowering PZR level. At 17% letdown isolates, and charging continues filling the PZR until the Hi PZR Lvl setpoint on 2/3 channels.This is calculated to take a total time of 4.4 hours. Note the failed "as is" channel will not trip bistables. All distracters are plausible based on examinees knowledge of system.

Ref: Pressurizer L-P section C-Abnormal Operations 3. Level Control Failures. L-P pg 50.

Question 20 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28276
User-Defined ID:	2012 NRC EXAM RO Q20
Cross Reference Number:	3D.OA-12-A
Topic:	2012 NRC Exam RO Question 20
Num Field 1:	RO 3.8
Num Field 2:	SRO 3.9
Text Field:	028AA1.01
Comments:	Modified question High Cog RO Level **028 Pressurizer (PZR) Level Control Malfunction** **AA1. Ability to operate and / or monitor the following as they apply to** **the Pressurizer Level Control Malfunctions:** **(CFR 41.7 / 45.5 / 45.6)** AA1.01 PZR level reactor protection bistables 3.8* 3.9

Associated objective(s):

Given a set of plant conditions or parameters indicating a malfunction in the Pressurizer Level Control System, DISCUSS the integrated plant response to the event/casualty with no operator action

A reactor startup is in progress at 100 cps in the Source Range.

- The control power fuse for NI Channel N-32 blows.

Which ONE of the following describes the NEXT action to be taken?

A. Raise reactor power greater than P-6, then block Source Range Hi Flux trip.

B. Verify all rod bottom lights lit.

C. Manually reinsert Control and Shutdown banks.

D. Place the LEVEL TRIP switch for the affected channel on 1PM07J in BYPASS.

Answer: B

Answer Explanation:

Meets K/A Requires knowledge of the power supplies to the source range instrumentation and the interrelationship between that and a loss of Source Range Instrumentation.

Below P-6 and a loss of power to Source Range N-32, the channel will fail high (trip) making the 1 out of 2 coincidence for a reactor trip.
Place the LEVEL TRIP switch for the affected channel on 1PM07J in BYPASS is procedurally driven but only if greater than P-6 power.
Raise reactor power greater than P-6, then block Source Range Hi Flux trip is similiar in action to loss of intermediate range detector making this a plausible distractor.
Manually reinserting all rods is a conservative action that the examinee may consider as correct making this a plausible distractor.

Ref: BAR 1-11-A2 for setpoints and 1BOA Inst-1 for distracters

Question 21 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28278
User-Defined ID:	2012 NRC EXAM RO Q21
Cross Reference Number:	T.OA10-03
Topic:	2012 NRC Exam RO Question #21
Num Field 1:	RO 2.7
Num Field 2:	SRO 3.1
Text Field:	032AK2.01
Comments:	Bank Question RO Level High Cog **032 Loss of Source Range Nuclear Instrumentation** **AK2. Knowledge of the interrelations between the Loss of Source Range** **Nuclear Instrumentation and the following:** **(CFR 41.7 / 45.7)** AK2.01 Power supplies, including proper switch positions 2.7* 3.1

Associated objective(s):

ANALYZE a given set of plant conditions and DETERMINE: The Plant Response for a Failed Nuclear Instrument

Given:

- Unit 1 is being shutdown.
- The Main Turbine was taken off-line a few minutes ago.
- Reactor Power is at 8% and being reduced by driving rods.
- All Power Range Nuclear Instruments currently read 8%.

Suddenly the reactor automatically trips.

Under these conditions, which of the following causes an automatic reactor trip?

A. Loss of Control Power to IR NI channel N35

B. Loss of Control Power to SR NI channel N31

C. Loss of Instrument Power to SR NI channel N32

D. Loss of Instrument Power to PR NI channel N41

Answer: A

Answer Explanation:

This meets the KA, because a rapid survey of the control room after the trip would reveal the annuciator in red as the first out, and it could be confirmed by the CP fuse on the IR drawer glowing.

With reactor power below 10%, P-10 is no longer met, and the Intermediate Range NI trips are no longer blocked. The loss of Control Power to the IR NI results in a Trip signal. At 8% power, P-6 is still preventing the Source Ranges from being powered, so any loss of control or instrument power to the SR NI will have no impact. The coincidence for a Power Range Trip is 2/4, so a single failure has no impact in this case.

All distractors are plausible based on the examinees understanding of the system.

Ref: BAR 1-11-B2 and Intermediate Range NI L-P pg. 14

Question 22 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28255
User-Defined ID:	2012 NRC EXAM RO Q22
Cross Reference Number:	T.OA10-13
Topic:	2012 NRC Exam RO Question 22
Num Field 1:	RO 3.0
Num Field 2:	SRO 3.1
Text Field:	033AA2.05
Comments:	New Question Hi Cog RO Level **033 Loss of Intermediate Range Nuclear Instrumentation** **AA2. Ability to determine and interpret the following as they apply to** **the Loss of Intermediate Range Nuclear Instrumentation: (CFR: 43.5 / 45.13)** AA2.05 Nature of abnormality, from rapid survey of control room data . . . . 3.0* 3.1?

Associated objective(s):

ANALYZE a given set of plant conditions and DETERMINE: The Plant Response for a Failed Nuclear Instrument

Given:

- Unit 2 is at 90% power, with the following:
- Rod Control in Automatic, with CBD at 221 steps.
- All other systems normally aligned.
- Control Bank Worth is 9 pcm/step
- MTC is 9 pcm/ degree F

A turbine ramp down was initiated due to a slow loss of condenser vacuum which causes a 3 degree change in Tave.

Exhaust Hood pressure meters (condenser vacuum) are ALL pegged HIGH on 1PM03J. (NO other operator actions are taken.)

Given the above, and neglecting any effects of Xenon, what is the final rod position for the controlling bank?

A. 0

B. 218

C. 223

D. 224

Answer: A

Answer Explanation:

Meets KA. Requires examinee to determine from conditions provided, what rod position is based on lowering condenser vacuum.

When Condenser vacuum degraded past 10”HgA, an automatic trip occurred allowing all the rods (including the controlling bank) to drop to 0 steps. If the examinee calculates necessary rod motion based on 3 degree change in Tave times 9pcm/degree and divides by 9pcm/step, he will get 3 steps on the controlling bank. 218 and 224 are 3 steps from 221. 223 is the automatic rod stop for C-11.

All distractor are plausible based on conditions stated and examinees knowledge.

Ref: BAR 2-18-E2

Question 23 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28256
User-Defined ID:	2012 NRC EXAM RO Q23
Cross Reference Number:	T.OA38-09
Topic:	2012 NRC Exam RO Question 23
Num Field 1:	RO 2.5
Num Field 2:	SRO 2.5
Text Field:	APE 051 AA1.04
Comments:	New Question Hi Cog RO Level **APE: 051 Loss of Condenser Vacuum** **AA1. Ability to operate and / or monitor the following as they apply to** **the Loss of Condenser Vacuum:** **(CFR 41.7 / 45.5 / 45.6)** AA1.04 Rod position 2.5* 2.5*

Associated objective(s):

ANALYZE a given set of conditions and DETERMINE the required operator actions to ensure acceptable turbine operation

Given the following plant conditions:

- Unit 1 operators have implemented 1BOA SEC-8, STEAM GENERATOR TUBE LEAK.
- 1CV121 has been fully opened at Step 1.a, "Maintain Pzr Level" and the crew is currently performing Step 1.b, "Check Pzr level - "STABLE OR RISING."
- Pzr level is LOWERING at 1% per 10 minutes.

For these conditions, and in accordance with 1BOA SEC-8, the operators will ensure letdown flow is…

A. 20 GPM via Excess Letdown.

B. 45 GPM to continue cleanup of the RCS and stabilize PZR level.

C. 75 GPM to continue cleanup of the RCS and stabilize PZR level.

D. 120 GPM to continue cleanup of the RCS.

Answer: C

Answer Explanation:

Meets KA. Requires examinee to compare letdown flows from "normal" to what is required during a SG Tube Leak procedure implementation. Letdown flows will affect VCT levels and hence make-up to the VCT and make-up to the RCS.

1BOA Sec-8, Step 1 b. RNO establishes 75 GPM letdown. This is to allow cleanup of the RCS while reducing RCS loss enough for charging flow to makeup for a small tube leak.
120 GPM is normal letdown rate.
Excess letdown is considered a plausible distracter based on having a reduced capacity (flowrate).
The above distractors are therefore all plausible.

Ref: 1BOA Sec-8, Step 1 b. RNO

Question 24 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	28067
User-Defined ID:	CERT 2010 RO 24
Cross Reference Number:	T.OA43-05
Topic:	Letdown flowrate during BOA Sec-8
Num Field 1:	3.1
Num Field 2:	3.3
Text Field:	037AK3.03
Comments:	Bank question, Low Cog, RO Level References 1BOA Sec-8, Rev 105 **037 Steam Generator (S/G) Tube Leak** **AK3. Knowledge of the reasons for the following responses as they apply to** **the Steam Generator Tube Leak:** **(CFR 41.5,41.10 / 45.6 / 45.13)** AK3.03 Comparison of makeup flow and letdown flow for various modes of operation 3.1 3.3

Associated objective(s):

Byron ILT 09-1 Cert Exam

Given:

- Unit 2 is at 100% power with all systems aligned for normal operation.
- Unit Auxiliary Transformer (UAT) 241-1 just tripped on the associated Sudden Pressure Relay.
- All systems have responded as designed.

What is the status of the Unit 2 reactor, and the Fire Protection System?

	Unit 2 Reactor	Fire Protection
A.	Tripped	actuated
B.	On line	actuated
C.	Tripped	NOT actuated
D.	On line	NOT actuated

Answer: A

Answer Explanation:

Meets K/A. Requires knowledge of Fire Protection/Fire Fighting and system setpoints, response and affect on the plant.
The Sudden Pressure actuation results in a unit transformer trip, which in turn trips the Main Generator, Turbine and Reactor (since >P-8). Sudden pressure is one of 2 transformer trips that also actuates a Fire Protection deluge on the transformer. This question is designed to test the examinee's knowledge of Fire Protection and the actuation affects on the plant. All distracters are plausible based on the misunderstanding of system interrelationships.

Ref: BAR 2-19-B5 and BAR 2-19-E2

Question 25 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28277
User-Defined ID:	2012 NRC EXAM RO Q25
Cross Reference Number:	S.FP1-04-A
Topic:	2012 NRC Exam RO Question #25
Num Field 1:	RO 3.1
Num Field 2:	SRO 3.9
Text Field:	067AK1.02
Comments:	New Question RO Level High Cog **067: Plant fire on site** **AK1. Knowledge of the operational implications of the following concepts as** **they apply to Plant Fire on Site:** **(CFR 41.8 / 41.10 / 45.3)** AK1.02 Fire fighting 3.1 3.9

Associated objective(s):

DISCUSS how the water fire protection subsystem header is kept pressurized. Include: Jockey Pumps

5 minutes ago a fire broke out in the Upper Cable Spreading Room.

- Both units were manually tripped from 100% power.
- BOA Pri-5, CONTROL ROOM INACCESSIBILITY, has been entered on both Units.
- An immediate control room evacuation was NOT required.
- The Unit-1 NSO and Unit Supervisor are at the Remote Shutdown Panel performing Attachment A, TRANSFER OF EQUIPMENT TO LOCAL CONTROL.

You are the Unit 1 Assist NSO and are currently in the Control Room observing the transfer of control.

What indications and controls will you have available to you in the Main Control Room for the running CV pump when the Unit NSO places the Remote/Local control switches in LOCAL at 1PL04/05J panels?

A. NO pump indicating lights AND NO pump control; Annunciator 1-9-E3, CHG PUMP/VLV LOCAL CONT is LIT.

B. Pump indicating lights BUT NO pump control; Annunciator 1-9-E3, CHG PUMP/VLV LOCAL CONT is LIT.

C. NO pump indication BUT pump control will be available.

D. Pump indication AND pump control are available; Annunciator 1-9-E3, CHG PUMP/VLV LOCAL CONT is LIT.

Answer: A

Answer Explanation:

Meets K/A. Requires examinee knowledge related to what control room indications and controls are lost when control is transferred to the RSP. In addition, must know what annunciators alarm when control is transferred.
When control is transferred to the RSP, no indication of pump operation is available in the control room (1PM05J), except for ammeter indication. No pump indicating lights will be available. This function is controlled by the Local/ Remote switch at the Remote SD panel (1PL04J). Control capability is also lost (refer to BAR 1-9-E3). An annunciator will sound, informing the operator that control is lost.
All distractors are plausible based on misconceptions of loss of either indication or control.

Ref: BAR 1-9-E3 and 6E-1-4030 CV01

Question 26 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28282
User-Defined ID:	2012 NRC EXAM RO Q26
Cross Reference Number:	3D.OA-27-B
Topic:	2012 NRC Exam RO Question #26
Num Field 1:	RO 4.2
Num Field 2:	SRO 4.4
Text Field:	068G2.2.44
Comments:	New Question Low Cog **068 Control Room Evacuation** 2.2.44 **Ability to interpret control room indications to verify the status and operation of** **| a system, and understand how operator actions and directives affect plant and** **| system conditions.** | (CFR: 41.5 / 43.5 / 45.12) IMPORTANCE RO 4.2 SRO 4.4

Associated objective(s):

Given a set of plant conditions or parameters indicating a Control Room Inaccessibility condition and a set of plant procedures, IDENTIFY the correct procedure(s) to be utilized and DESCRIBE required operator actions

In order to enter 1BEP ES-0.0, "REDIAGNOSIS" which of the following conditions is REQUIRED to be met?

A. Reactor Trip has occurred but NO Safety Injection has been actuated.

B. The last step of 1BEP-0, "REACTOR TRIP OR SAFETY INJECTION" has been reached with NO procedure transition identified.

C. 1BEP-0, "REACTOR TRIP OR SAFETY INJECTION" has been implemented AND the crew has transitioned to another Emergency Procedure.

D. Reactor Trip has occurred AND Shift Manager permission is received.

Answer: C

Answer Explanation:

This meets the K/A because the candidate must determine the required condition of the plant in order to use the rediagnosis procedure which is an operational implication. According to procedure entry conditions statement 1BEP-0 must have been implemented and exited to use 1BEP ES-0.0.
Distractors are plausible because they all describe possible plant conditions that could readily enter into a decision to use BEP ES-0.0, or because SM permission is a likely prerequisite before this unusual procedure is used after normal diagnostic steps have been inadequate.

ref: 1BEP ES-0.0

Question 27 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	28157
User-Defined ID:	BYLI-EP0-107
Cross Reference Number:	T.EP01-06-B
Topic:	Rediagnosis entry
Num Field 1:	3.4
Num Field 2:	4.0
Text Field:	WE01EK1.2
Comments:	Bank Question, Low Cog, RO Level **E01 Rediagnosis** **EK1. Knowledge of the operational implications of the following concepts as** **they apply to the (Reactor Trip or Safety Injection/Rediagnosis)** (CFR: 41.8 / 41.10 / 45.3) EK1.2 Normal, abnormal and emergency operating procedures associated with (Reactor Trip or Safety Injection / Rediagnosis). IMPORTANCE RO 3.4 SRO 4.0 Reference: 1BEP ES-0.0, Rediagnosis

Associated objective(s):

GIVEN a set of plant conditions, DIAGNOSE and ANALYZE a Rediagnosis

If an INADVERTANT Phase B Isolation occurs, how are the Reactor Coolant Pumps affected?

A. Thermal barrier CC flow is RAISED.

B. Motor bearing cooling flow is ISOLATED.

C. Seal water injection flow is LOWERED.

D. Seal water return flow is ISOLATED.

Answer: B

Answer Explanation:

Meets K/A. Requires examinee to have knowledge of cooling flow isolation to RCP motor bearings from a Phase B containment isolation signal.

All distractors are plausible because they are interconnected with the RCP.
Seal water return flow is isolated on a Phase A signal not Phase B.
Seal Injection flow is affected by SI initiation but not by a Phase B.
Thermal barrier flow is isolated on a Phase B so flow would lower rather than raise.

Ref: BOP RC-1 E.5 (pg 8) and BEP-0 Operator Action Summary

Question 28 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28337
User-Defined ID:	2012 NRC EXAM RO Q28
Cross Reference Number:	S.RC2-09-E
Topic:	2012 NRC Exam RO Question 28
Num Field 1:	RO 2.7
Num Field 2:	SRO 3.0
Text Field:	003K1.08
Comments:	New Question, Low Cog, RO Level **003 Reactor Coolant Pump System (RCPS)** **K1 Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems:** (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.08 Containment isolation 2.7* 3.0*

Associated objective(s):

DESCRIBE the operational limitations, and the reasons for them, on the Reactor Coolant Pump regarding: The Conditions That Dictate a Trip of the Reactor Coolant Pump by the Operator

Given:

? Unit 2 is in Mode 1 at 100%.
? PZR level is stable and at design level for this power.
? Letdown line flow = 75 gpm.
? 2CV8149C, 75 GPM LTDWN ORIF 2C ISOL VLV, is open.
? Charging header pressure is 2300 psig.

Due to an electrical malfunction, 2CV8149B, 75 GPM LTDWN ORIF 2B ISOL VLV, fails OPEN.

Which ONE (1) of the following statements describes the affects on the CVCS design considerations resulting from this malfunction 10 minutes AFTER THE FAILURE?

This lineup results in exceeding.....

A. the design flowrate of the Mixed Bed Demineralizer resulting in channeling. Demineralizer Decontamination Factor (DF) will lower.

B. the make-up capacity of a single charging pump. Pressurizer level lowers to the Letdown isolation setpoint.

C. the pressure control capability of 2CV131, LTDWN LINE PRESS CONT VLV. Letdown line pressure will rise causing continuous lifting of the letdown relief valve.

D. the capacity of LTDWN HX OUT TEMP CONTROL, 2CC130A. Letdown temperature will rise, causing CV pump suction temperature to be exceeded.

Answer: A

Answer Explanation:

Meets K/A. Requires examinee knowledge of CVCS design parameters and normal system operation.

Letdown line pressure will momentarily be affected by increased flow, however the increase in flow is well within the capability of 2CV131 to control pressure.
Letdown line temperature will momentarily be affected by increased flow, however the increase in flow is well within the capability of 2CV130A to control temperature.
Pressurizer level will momentarily drop but 1CV121 will open to maintain level. Increased flow is within the capability of a single charging pump at this pressure, an alarm will sound, however.
Demin flow exceeding 120 gpm is a design setpoint at which above, channeling may occur in the demin. This will cause demineralizer efficiency to lower which will lower the Decontamination Factor.

Ref: BAR 1-8-A5, BOP CV-9 E.1 and CVCS L-P page 8 d.1.e) on max flow. Requires the examinee to know that channeling will cause DF to lower.

Question 29 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28287
User-Defined ID:	2012 NRC EXAM RO Q29
Cross Reference Number:	S.CV1-10-A
Topic:	2012 NRC Exam RO Question 29
Num Field 1:	RO 2.6
Num Field 2:	SRO 2.9
Text Field:	004K6.22
Comments:	New question, Lo Cog, RO Level **004 Chemical and Volume Control System (CVCS)** **K6 Knowledge of the effect of a loss or malfunction on the following CVCS** **components:** (CFR: 41.7 / 45.7) K6.22 Design minimum and maximum flow rates for letdown system. 2.6 2.9

Associated objective(s):

EXPLAIN how the CVCS letdown flowrate is controlled at: normal RCS operating pressure

Given the following sequence of events:

- Unit 1 was at 100% power, normal alignment.
- The Unit experiences a LOCA with an AUTO Safety Injection.
- 1A RH pump tripped and will NOT re-start.
- All other equipment functioned as designed.

The crew has just completed ALL steps of 1BEP ES-1.3 “Transfer to Cold Leg Recirculation Alignment".

How many CV pumps will be running and where is/are the CV pump(s) suction source?

	number of CV pp(s) running	suction source
A.	1	directly from the recirc sump
B.	2	directly from the recirc sump
C.	1	1B RH pump discharge
D.	2	1B RH pump discharge

Answer: D

Answer Explanation:

Question meets K/A - requires examinee knowledge of how 1A RH pump malfunction will effect ECCS system (CV pps).
Per 1BEP ES 1.3, if 1A RH pp not running, step for opening 1CV8804A is skipped. Therefore CV pp suction is supplied by crosstie to SI pumps only. The 1SI8924 is normally open and 1SI8804B along with 1SI8807A/B would be opened by the procedure. Both CV pumps remain running.
Choice A is incorrect, both CV pumps will be running. Suction from the recirc sump is incorrect.
Choice B is incorrect, the pumps will be taking a suction from the B RH pp.
Choice C is incorrect, 1 RH pump can supply both RH pumps
Choice D is correct, see explanation above

Ref: 1BEP ES-1.3 step 5 (pg 9)

Question 30 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28270
User-Defined ID:	2012 NRC EXAM RO Q30
Cross Reference Number:	3D.EP-14-B
Topic:	2012 NRC Exam RO Question 30
Num Field 1:	RO 3.7
Num Field 2:	SRO 3.8
Text Field:	005K3.05
Comments:	New Question High Cog RO Level **005 Residual Heat Removal System (RHRS)** **K3 Knowledge of the effect that a loss or malfunction of the RHRS will** **have on the following:** (CFR: 41.7 / 45.6) K3.05 ECCS 3.7* 3.8*

Associated objective(s):

Given a set of plant conditions or parameters indicating a need to Transfer to Cold Leg Recirc, and a set of plant procedures, IDENTIFY the correct procedure(s) to be utilized and DISCUSS required operator actions

Given:

- Unit 1 was at 100% power, all systems normally aligned.
- A Large Break LOCA occurred, followed by an immediate Loss of Off-Site Power.
- TWO (2) SI accumulators failed to inject.

The effect of the failure of the SI Accumulators to inject is…

A. insufficient water will be available during the blow down and refill phases of the LOCA.

B. Containment Recirculation Sump pH will be lower than required.

C. Containment Recirculation Sump inventory will be insufficient for long-term core cooling.

D. less nitrogen gas available to block natural circulation.

Answer: A

Answer Explanation:

Question meets KA - question requires examinee knowledge of the effect of a loss SI accumulators will have on the ECCS.
A is correct, T.S. 3.5.1 Bases states the function of ECCS accumulators are to supply water to the reactor vessel during the blowdown phase of a LOCA and to provide inventory to assist in the refill phase.
B is incorrect, containment recirc sump pH would be slightly higher than normal due to less boric acid in sump.
C is incorrect, accumulators do not contribute enough water to containment recirc sump to affect long term core cooling
D is incorrect, natural circulation is not possible for a LBLOCA, reflux boiling is.

Ref. ECCS L-P Pg. 29 of 78 refers to T.S. and Bases
and Mitigating Core damage L-P

Question 31 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28267
User-Defined ID:	2012 NRC EXAM RO Q31
Cross Reference Number:	S.EC1-02
Topic:	2012 NRC Exam RO Question 31
Num Field 1:	RO 3.4
Num Field 2:	SRO 3.9
Text Field:	006K6.02
Comments:	From 2009 Braidwood Cert. Exam Low Cog RO Level **006 Emergency Core Cooling System (ECCS)** **K6 Knowledge of the effect of a loss or malfunction on the following** **will have on the ECCS:** (CFR: 41.7 / 45.7) K6.02 Core flood tanks (accumulators) 3.4 3.9

Associated objective(s):

EXPLAIN the design criteria per 10CFR50.46 for Emergency Core Cooling Systems

Unit 1 is in MODE 1 with the following initial conditions in the PRT:

- PRT level = 71%
- PRT pressure = 4.5 psig
- PRT temperature = 85°F

A Pressurizer PORV inadvertently OPENED and has since been CLOSED.

Conditions in the PRT are now:

- PRT level = 77% and stable
- PRT pressure = 7.2 psig and stable
- PRT temperature = 102°F and stable

Based on these conditions, what is the NEXT required action?

A. Drain the PRT to reduce pressure.

B. Reduce temperature in the PRT.

C. Vent the PRT to reduce pressure.

D. Inert the PRT to limit the H2 concentration.

Answer: A

Answer Explanation:

Meets K/A. Utilizes procedural actions to mitigate overpressurization of the PRT and subsequent overpressurization of the waste gas vent header. Futhermore it requires the examinee to predict which parameter is off-normal from pressure/temperature or level. At 6 psig, RY469 auto closes to GW from the PRT. The BAR says probable cause is (1) Valve leakoff or relief valve flow, (2) PORV or Safety Valve lifted, (3) Filling PRT, and (4) N2 Regulator failure. The PRT will be drained first, which will lower PRT pressure, before the vent valve can be re-opened.
Requires examinee to understand that the PRT is auto isolated from the waste gas header on an overpressure event. Also requires knowledge of normal/abnormal conditions to assess what actions need to be taken. Opening 1RY469 only will not occur until PRT pressure is reduced to less than 6 psig. The BAR stipulates to ensure the N2 supply to the PRT is isolated along with the PW supply.
1RY8033 is an AOV but it has a pressure regulating valve downstream (1RY8034 set at 3 psig) so under the pressure as stated in the stem this action would have no effect.
Reducing temperature is plausible but the Hi Temp. alarm has not annunciated as the setpoint is 115 degrees.

Ref: BAR 1-12-B7

Question 32 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28288
User-Defined ID:	2012 NRC EXAM RO Q32
Cross Reference Number:	S.RY1-13
Topic:	2012 NRC Exam RO question #32
Num Field 1:	RO 2.5
Num Field 2:	SRO 2.9
Text Field:	007A2.04
Comments:	New Q High Cog **007 Pressurizer Relief Tank/Quench Tank System (PRTS)** **Ability to (a) predict the impacts of the following malfunctions or operations** **on the P S; and (b) based on those predictions, use procedures to correct,** **control, or mitigate the consequences of those malfunctions or operations:** (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.04 Overpressurization of the waste gas vent header 2.5 2.9

Associated objective(s):

STATE the normal operating pressure, temperature, and level of the PRT

Given:

- Unit 1 was just shutdown for a refueling outage after 500 days of continuous operation.
- CC and SX is aligned to provide maximum cooling.
- At 0650 RH is placed in shutdown cooling
- At 0700 CC Hx outlet temperature reaches 100 degrees F and is RISING at 1 degree F per minute.

Assume the rate of temperature change is constant.

When will the CC HX outlet temperature REACH the extended temperature limit?

A. 0705

B. 0720

C. 1005

D. 1020

Answer: B

Answer Explanation:

Meets K/A. Requires knowledge of CC system design temperature.

Per a statement in BOP CC-1 and T.S. Bases of 3.7.7 (pg.3), the normal design temperature of CC is 105 degrees. Temperature may be allowed to go to 120 degrees not to exceed a 3 hour time frame.
The correct answer is correct based on 20 minutes time to reach 120 degrees.
0705 is plausible as at 0705 we will reach the normal design limit of 105 degrees.
1005 is plausible based on the 3 hour time limit after reaching 105 degrees but you would be greater than 120 degrees which is not permissible.
1020 is plausible based on the same logic used above accept that this includes the time necessary to reach 120 degrees.

Ref. BOP CC-1 D. Precautions, 4.

Question 33 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28289
User-Defined ID:	2012 NRC EXAM RO Q33
Cross Reference Number:	3C.RH-04-A
Topic:	2012 NRC Exam RO question #33
Num Field 1:	RO 2.9
Num Field 2:	SRO 3.1
Text Field:	008A1.02
Comments:	New Question, Lo cog **008 Component Cooling Water System (CCWS)** **Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including**: (CFR: 41.5 / 45.5) A1.02 CCW temperature 2.9 3.1

Associated objective(s):

DESCRIBE plant critical parameters and system response to using the RH Heat Exchanger including effects on the CC System

Given the following plant conditions on Unit 1:

- A manual reactor trip and SI have been actuated.
- A PZR safety valve has been stuck slightly open for ten minutes.
- RCS pressure is 1910 psig and slowly dropping.
- PZR vapor space temperature is 630 °F and stable.
- PRT level is 80% and slowly rising.
- PRT pressure is 35 psig and slowly rising.
- The open PZR safety valve tailpipe temperature is 390 °F and stable.

The NSO believes there is a problem with the safety valve tailpipe temperature since it is NOT reading as expected for the current plant conditions.

Currently, the safety valve tailpipe temperature indication should be reading ...

A. 257 °F.

B. 281 °F.

C. 390 °F.

D. 630 °F.

Answer: B

Answer Explanation:

Meets K/A. Requires examinee to utilize the Steam Tables for an isenthalpic process involving leakage past a PZR Safety valve.

281 °F is the correct answer for 35 psig or 50 psia
257 °F. is incorrect. That is the temperature for 35 psia.
630 °F is incorrect but considered plausible because that is the temperature of the fluid as it started the throttling process.
390 °F is incorrect but considered plausible because it is included in the question stem

Ref: Steam Tables

Question 34 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	4
Difficulty:	0.00
System ID:	26488
User-Defined ID:	BYLCATH3001
Cross Reference Number:	A.TH3-09
Topic:	Stuck Open Safety Tailpipe Temp
Num Field 1:	2.6
Num Field 2:	3.0
Text Field:	010000K5.02
Comments:	Created from BWLCATH3001 Ref: Steam Tables Last reviewed: 08/11 PRA Item: N OPEX: TMI Bank Question, High Cog. **010 Pressurizer Pressure Control System** **K5 Knowledge of the operational implications of the following concepts as** **the apply to the PZR PCS:** (CFR: 41.5 / 45.7) K5.02 Constant enthalpy expansion through a valve 2.6 3.0*

Question 34 Table-Item Links

LORT Question References

K A #1 - 010000K501RO - 3.5 SRO - 4.0

Associated objective(s):

EXPLAIN the Thermodynamic effects of a throttling process

Given:

- The unit is at full power in a normal full power line-up
- 1PT-457 has failed LOW.
- Control Room actions associated with the failed channel have been taken.
- Instead of the bistables for PT-457, ASSUME the following bistables for A Loop NR RTD are locally tripped IN THE ORDER LISTED.
  - ➢ OPDT Trip
  - ➢ OPDT Runback
  - ➢ OTDT Trip
  - ➢ OTDT Runback

Which of the selections below identify the bistable and procedure necessary to address the control or protection action?

	At the time of tripping bistable	The procedure to enter is:
A.	OPDT Trip	1BEP-0, REACTOR TRIP OR SAFETY INJECTION
B.	OPDT Runback	BOP CV-6, OPERATION OF THE REACTOR MAKEUP SYSTEM IN THE BORATE MODE
C.	OTDT Trip	1BEP-0, REACTOR TRIP OR SAFETY INJECTION
D.	OTDT Runback	BOP CV-6, OPERATION OF THE REACTOR MAKEUP SYSTEM IN THE BORATE MODE

Answer: C

Answer Explanation:

Meets KA. Requires examinee to predict the impact of improper tripping of bistables associated with the RPS and to select the procedure that will mitigate the consequences of that action. Also requires examinee to know what bistables trip without operator action due to 1PT457 failing low. They are: OTDT(TD 431C),OTDT/C-3 (TD431D) PZR Lo Press(PB457C), PZR Lo Press SI(PB457D), and PZR Press below P-11(PB457B)

The correct answer is correct because a 2 out of 4 coincidence will be made up on OTDT requiring entry into the reactor trip response procedure.
All distracters are plausible if the examinee mistakenly feels OPDT will be affected or the runback circuits will be made up.
Runback will cause rods to drive in, if Lo-1 or Lo-2 RIL is reached, boration will be procedurally directed.

Ref: BOA Inst. 2 Attachment A for failed NR RTD
Attachment B for failed PZR Press. channel

Question 35 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28361
User-Defined ID:	2012 NRC EXAM RO Q35
Cross Reference Number:	T.OA11-25
Topic:	2012 NRC Exam RO Question 35
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	New Question, High Cog, RO level **012 Reactor Protection System** **A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:** (CFR: 41.5 / 43.5 / 45.3 / 45.5) A2.03 Incorrect channel bypassing 3.4 3.7

Associated objective(s):

DESCRIBE the expected plant response for the failures listed in TKO T.OA11-03 thru T.OA11-20 including: Alarm Status Changed, Automatic Actions That Happen and Controls and Permissives That are Affected

What is the (1) reason that 1 CV pump and both SI pumps are taken out of service during a plant cooldown to refueling conditions AND (2) when must that requirement be met?

	(1) Reason	(2) When
A.	These components are no longer needed to mitigate accident conditions.	between 330 and 350 degrees
B.	Ensures the mass addition can be relieved by a single PORV or RH Suction Relief valve.	prior to MODE 4 entry
C.	Ensures the mass addition can be relieved by a single PORV or RH Suction Relief valve	between 330 and 350 degrees
D.	These components are no longer needed to mitigate accident conditions.	prior to MODE 4 entry

Answer: C

Answer Explanation:

Meets K/A. Requires examinee to know the reason for opening the breaker on the High Head injection pumps prior to lowering temperature below 330. The design feature portion of the KA is addressed by the capability of the system to remove the postulated mass addition that may be caused. The reason is stated in the L-P and has an RO objective to support it and in the bases section of Tech Spec. All other distractors are plausible depending on examinees knowledge. Prior to MODE 4 entry is incorrect however plausible because prior to mode 4 entry means mode 3. 2 ECCS subsystems are required by TS in MODE 3. Components no longer needed to mitigate accident conditions is considered plausible.

Ref: BGP 100-5 step 38
BGP 100-5 L-P pg 9

Question 36 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28291
User-Defined ID:	2012 NRC EXAM RO Q36
Cross Reference Number:	3C.GP-05-B
Topic:	2012 NRC Exam RO Question 36
Num Field 1:	RO 3.0
Num Field 2:	SR0 3.4
Text Field:	013K4.19
Comments:	New Question Low Cog. **013 Engineered Safety Features Actuation System (ESFAS)** **K4 Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following**: (CFR: 41.7) K4.19 Reason for opening breaker on high-head injection pump . . . . 3.0* 3.4* Ref.: T/S 3.4.12 as bases

Associated objective(s):

Given the appropriate section of Tech Specs and a set of plant conditions or parameters indicating a possible LCO violation due to the Cold Overpressure Protection System, DETERMINE Tech Spec compliance and required action

DESCRIBE the bases for the CVCS Tech Spec operability requirements.

37 ID: 2012 NRC EXAM RO Q37 Points: 1.00

What does the box symbol around a procedure step number of an E-series procedure represent?

It represents a/an...

A. Continuous Action Summary step.

B. Immediate Action step.

C. Operator Action Summary step.

D. Sequencing step where all sub-steps must be completed.

Answer: B

Answer Explanation:

Meets KA. Requires examinee knowledge of EOP symbols used during the implementation of procedures which are used to mitigate the conditions associated with ESFAS actuations.

The correct answer is correct based on BAP 1310-10 pg 15..

A continuous action summary step, operator action summary step or a sequencing step are all included in the same procedure making these distractors plausible.

Ref: BAP 1310-10 pg 15

Question 37 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28356
User-Defined ID:	2012 NRC EXAM RO Q37
Cross Reference Number:	T.AM04-02
Topic:	2012 NRC Exam RO Question 37
Num Field 1:	RO 3.4
Num Field 2:	SRO 4.1
Text Field:	013gen. 2.4.19
Comments:	New Question, Low Cog, RO level **013 Engineered Safety Features Actuation System (ESFAS)** 2.4.19 **Knowledge of EOP layout, symbols, and icons.** | (CFR: 41.10 / 45.13) IMPORTANCE RO 3.4 SRO 4.1

Associated objective(s):

DESCRIBE the 2-column format used in the Abnormal and Emergency operating procedures

38	ID: 2012 NRC EXAM RO Q38	Points: 1.00

Given:

- Unit 1 is operating at full power in a normal full power alignment.
- 1A Containment Chiller is running.
- A fault occurs on SAT 142-1.
- NO operator actions are taken in response to the fault.

What is the status of Unit 1A Containment Chiller and where is it powered from following the fault?

	status	powered from
A.	running	UAT 141-1: 4KV winding
B.	running	UAT 141-2 : 4KV winding
C.	running	1A DG
D.	tripped	UAT 141-1 : 4KV winding to Unit Sub-station 133Y

Answer: A

Answer Explanation:

Meets K/A. Requires examinee to know power supply of the containment chillers and in addition how those power supplies are affected by a System Aux. Transformer fault.

The 1A Containment Chiller is powered from bus 143 which is powered from the Unit Aux. Transformer (141-1) when the unit is above approximately 10% power.
The distractors are all plausible based on: They are all fed from 4KV buses. 1 is ESF, 1 is a Non-ESF bus and the incorrect division (141-2) and 1 is the correct division but the incorrect voltage level (U.S.S. 133Y)
If the examinee feels the chiller will trip due to an ABT action, makes the "tripped" portion of the distractor plausible.

Ref: BGP 100-3 step 43 (pg. 51) which is performed after reaching 160MWe
Horse Note AC-3

Question 38 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28293
User-Defined ID:	2012 NRC EXAM RO Q38
Cross Reference Number:	S.VP1-13
Topic:	2012 NRC Exam RO Question 38
Num Field 1:	RO 2.5
Num Field 2:	SRO 2.4
Text Field:	022K2.02
Comments:	New Question, High Cog. **022 Containment Cooling System (CCS)** **K2 Knowledge of power supplies to the following:** (CFR: 41.7) K2.02 Chillers 2.5* 2.4*

Associated objective(s):

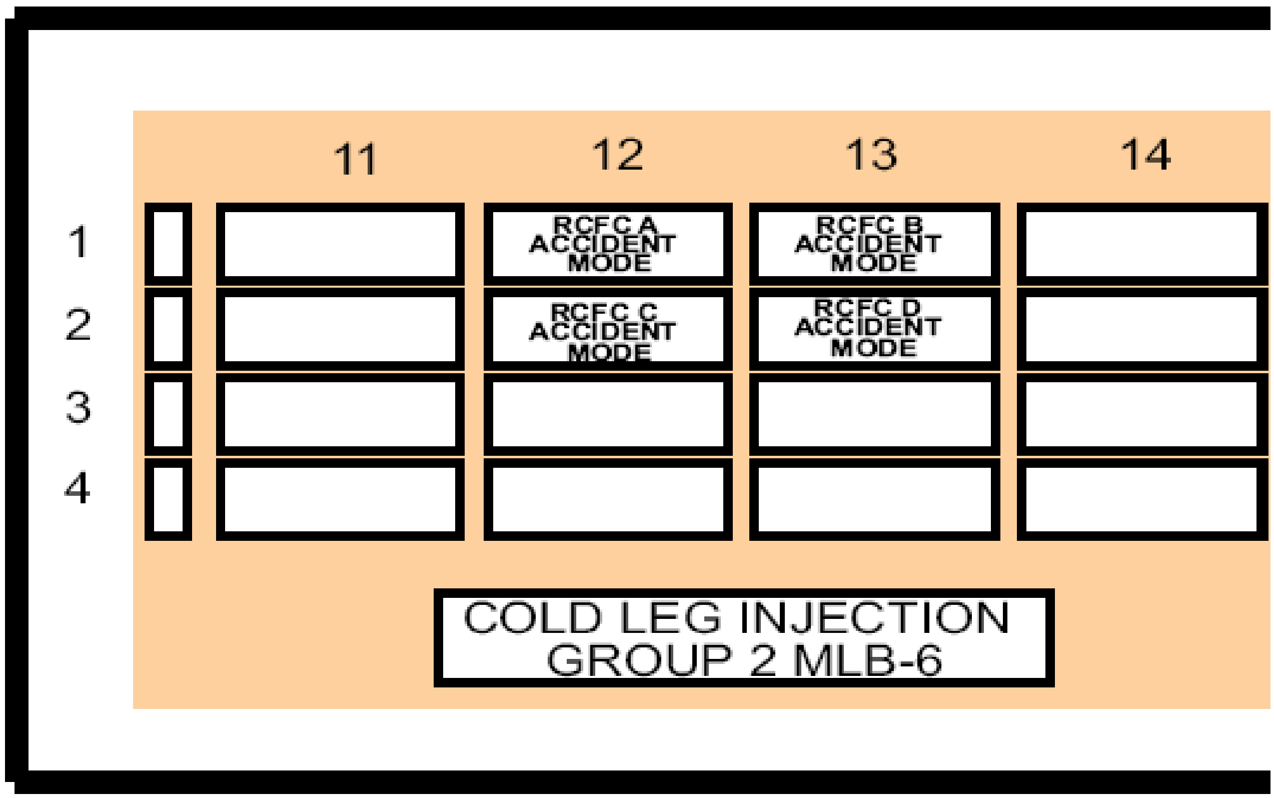
Given a set of plant conditions, PREDICT the impact, on the plant and/or Containment ventilation system, of various failures, including instrument malfunctions, electrical supply failures and component malfunctions

Given:

- Unit 1 was at full power in a normal full power line-up
- A Large LOCA has occurred.
- NO manual Control Switch operations have been taken by the crew.
- All Safety Systems responded as designed EXCEPT:

➢ The status light at location 12-2 is DARK (depicted below).

From the below list, which selection is the reason the indicator did NOT light?



A. 1WO006A, 1A and 1C CHILL WTR INLET CNMT ISOL VLV is OPEN

B. C RCFC running in HIGH speed.

C. 1C RCFC Vibration alarm has annunciated.

D. 1SX027A, 1A and 1C SX OUTLET VLV, is OPEN.

Answer: B

Answer Explanation:

Meets K/A. Requires examinee to have knowledge of Control Board Indications of the Containment Cooling System.

To obtain the light indication requires:
RCFC running in slow speed (specific to the individual RCFC)
Common to 1A and 1C RCFCs are:
SX aligned to the RCFC (1SX016A and 1SX027A) open

SX bypass open around the 1A Containment Chiller (1SX147A)
SX Isolated to the 1A Containment Chiller (1SX112A and 114A) closed

All other distractor are plausible because the are tied into the Containment Cooling function.
The High Vibration brings in an alarm only, no auto trip function

Ref: 6E-1-4030AN057
BAR 1-3-C5

Question 39 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28294
User-Defined ID:	2012 NRC EXAM RO Q39
Cross Reference Number:	S.VP1-06
Topic:	2012 NRC Exam RO Question 39
Num Field 1:	RO 4.2
Num Field 2:	SRO 4.1
Text Field:	022G2.4.31
Comments:	New Question, High Cog, RO level **022 Containment Cooling System (CCS)** 2.4.31 **Knowledge of annunciator alarms, indications, or response procedures**. | (CFR: 41.10 / 45.3) | IMPORTANCE RO 4.2 SRO 4.1

Associated objective(s):

DESCRIBE the operation of the reactor containment fan coolers during normal operation. Compare/Contrast this with operation during a Loss of Coolant Accident conditions

Given:

- The unit was at rated power.
- A Large Break LOCA has occurred.
- All systems responded as designed.
- Containment pressure peaked at 23 psig.
- Containment pressure is currently at 4 psig.
- ECCS pumps were swapped over to recirculation mode at 46.7% level in the RWST.
- There are currently 180,000 gallons in the RWST.

How long can you operate in this configuration before you are procedurally directed to swap CS pump suctions to the Containment Recirc. Sumps?

The procedurally directed swap will be required in____ minutes.

A. less than 12

B. 13 - 24

C. 25 - 36

D. greater than 36

Answer: B

Answer Explanation:

Meets KA. Requires examinee to have knowledge of the interface between ECCS and Containment Spray and how long the RWST can be used as a suction source for the CS pumps.

Current RWST available is 180,000 gallons. Procedural swap for CS pumps occurs at 12%. 458,000 gallons X .12= 54,960 gallons.
Pump delivery is rated at 3,415 (A) and 3,925 (B) for a total of 7340 gpm.
180,000gals-54,960gals= 125,040 gallon until 12% is reached in RWST.
125,040gal/7340gpm=17.0 minutes

All other distractors are plausible based upon examinees knowledge of pump capacities and procedural requirements.

Ref: 1BEP ES-1.3 for when to swap CS pumps (12%) per step 9
CS pump capacity is given in CS L-P page 4
RWST level/capacity graph is in BTC 1.34

Question 40 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28338
User-Defined ID:	2012 NRC EXAM RO Q40
Cross Reference Number:	S.CS1-11-D
Topic:	2012 NRC Exam RO question 40
Num Field 1:	RO 4.2
Num Field 2:	SRO 4.2
Text Field:	026K1.01
Comments:	New Question, High Cog, RO Level **026 Containment Spray System (CSS)** **K1 Knowledge of the physical connections and/or causeeffect relationships between the CSS and the following systems:** (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.01 ECCS 4.2 4.2

Associated objective(s):

DISCUSS the lineup and operation of the CS System during the following modes of operation:
Containment Phase B Isolation.

Given:

- Unit 2 was at full power when a Steam Line Break occurred in the containment.
- Containment pressure is currently at 28 psig and RISING
- The automatic start signal for the CS pumps failed to start the pumps.
- You are in 2BEP-0, REACTOR TRIP OR SAFETY INJECTION Attachment C, MANUAL CS ACTUATION attempting to manually start the CS pumps.

Immediately after OPENING which of the following valves, will the CS pump run light be LIT?

A. 2CS001A/B, PP 2_ RWST SUCT VLV.

B. 2CS007A/B, PP 2_ HDR ISOL VLV.

C. 2CS010A/B, EDUC 2_ INLET FLOW CONT VLV.

D. 2CS019A/B, 2_ SPRAY ADD VLV.

Answer: D

Answer Explanation:

Meets KA. Requires examinee to operate and or monitor CS control switches and monitor the effects on the CS system.

The CS pumps start on a HI-3 cont. pressure when 1CS019A is opened.
All distractors are plausible because they are listed steps in Att. C of EP-0

Ref: 6E-1-4030CS01

Question 41 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28339
User-Defined ID:	2012 NRC EXAM RO Q41
Cross Reference Number:	S.CS1-07
Topic:	2012 NRC Exam RO Question 41
Num Field 1:	RO 4.5
Num Field 2:	SRO 4.3
Text Field:	026A4.01
Comments:	New Question, Low Cog. RO Level **026 Containment Spray System (CSS)** **A4 Ability to manually operate and/or monitor in the control room:** (CFR: 41.7 / 45.5 to 45.8) A4.01 CSS controls . . . . 4.5 4.3 ref: 4030CS-1

Associated objective(s):

DESCRIBE all controls and indications available on the main control boards for remote operation of the CS System

42	ID: 2012 NRC EXAM RO Q42	Points: 1.00

Given the following plant conditions on Unit 1:

- A reactor startup is in progress.
- The reactor is at the POAH with the plant in a normal lineup at the EOL.
- The BOP Operator is making preparations to start up the "B" Feedwater pump.
- 1PT-507, S/G Header Pressure, fails to 1200 psig.

In response to the 1PT-507 failure, the steam dumps will ...

A. open fully, but will close when actual steam pressure reaches 1092 psig.

B. open fully, but will close when T_{AVE} reaches 550 °F.

C. remain in their current position due to the steam dump controller being in Tave mode with Reactor Trip Breakers closed and Tave NOT above setpoint.

D. remain in their current position due to the steam dump controller being in the Tave mode with Rx Trip Breakers closed but no Load Rejection signal present.

Answer: B

Answer Explanation:

Meets K/A, requires examinee knowledge to control steam header pressure in the steam pressure mode, including protective feature that occurs when Tave limits are reached. Requires examinee to know that during a power ascension steam dumps will be in the steam pressure mode and will not transfer to the Tave mode until approximately 10% power.

Normal steam pressure setpoint of 1092psig. When controller fails to 1200 psig, will cause dumps to open fully until they are closed by Tave dropping to 550 degrees (P12).

All distractors are plausible based on the examinee's misconceptions of when we are in steam pressure or Tave modes of operation or if there is a separate input steam pressure signal other than PT 507.

Ref: Steam Dump L-P pg. 22

Question 42 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	0.00
System ID:	18042
User-Defined ID:	BWLC3CDU1014
Cross Reference Number:	3C.DU-01-A-2
Topic:	BWLC3CDU1014
Num Field 1:	RO 2.9
Num Field 2:	SRO 3.1
Text Field:	039000K4.04
Comments:	Bank Question, High Cog Ref: 4030 MS09, BGP 100-3 Last reviewed: 08/11 **039 Main and Reheat Steam System (MRSS)** **K4 Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following:** (CFR: 41.7) K4.04 Utilization of steam pressure program control when steam dumping through atmospheric relief/dump valves, including T-ave. limits . . 2.9 3.1

Associated objective(s):

Given a set of plant conditions, DETERMINE the proper response of the Steam Dump System in the following modes of operation:
Steam Pressure Mode.

Given the following conditions on Unit 1.

- The reactor is at 20% power.
- Steam Dump Mode Select Switch is in the Tave position.
- Rod Control is in Manual.
- The main turbine has just tripped.

With NO operator action, how many groups of steam dump valves will indicate OPEN (fully and/or partially) on 1PM02J, 1 minute after the event?

A. 1

B. 2

C. 3

D. 4

Answer: B

Answer Explanation:

Meets K/A, Requires examinee to monitor steam dump valves from the control board for proper operation.

Total capacity of steam dumps is 40% rated steam flow. Each of 4 groups therefore is capable of 10% steam flow. We would be on the load reject controller since RTB is closed. RCS temp (Tave) at 20% power is 557+ .2 (30) = 563. The load reject controller has a 3 degree deadband (560). Then for each 1 degree above 560 the output of the controller increases by 9.43%. 9.43% per degree times 3 degrees equals 28.29%controller output. Each group is opened at 25% increments, therefore the first group is full open and the second group is opening.
The distractor are plausible for any misconceptions the examinee has with the system or whether or not they believe the unit trips.

Ref: Steam Dump L-P pg 7-10 and
Horsenote MS4

Question 43 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28340
User-Defined ID:	2012 NRC EXAM RO Q43
Cross Reference Number:	3C.DU-01-A-1
Topic:	2012 NRC Exam RO Question 43
Num Field 1:	RO 2.8
Num Field 2:	SRO 2.9
Text Field:	039A4.07
Comments:	New Question, High Cog, RO level **039 Main and Reheat Steam System (MRSS)** **A4 Ability to manually operate and/or monitor in the control room:** (CFR: 41.7 / 45.5 to 45.8) A4.07 Steam dump valves. 2.8* 2.9

Associated objective(s):

Given a set of plant conditions, DETERMINE the proper response of the Steam Dump System in the following modes of operation:
Tave Mode.

Of the following, what is the MAXIMUM power level the 1A Motor Driven Feedwater pump can maintain SG levels using the FWRV bypass valves (1FW510A/520A/530A and 540A), per 1BGP 100-3, POWER ASCENSION,

A. 10%

B. 20%.

C. 30%

D. 40%

Answer: B

Answer Explanation:

Meets K/A. Requires examinee to have knowledge of reactor power limits associated with 1 feedwater pump operation and FW Bypass valve limitations

Ref: 1BGP-3 step 40 (pg. 48)

Question 44 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	3.00
System ID:	28034
User-Defined ID:	2012 NRC EXAM Q44
Cross Reference Number:	S.CD1-07-A
Topic:	Power allowed on FWRV bypass valves
Num Field 1:	2.7
Num Field 2:	2.9
Text Field:	059A1.03
Comments:	Bank Question, Low Cog, RO level **059 Main Feedwater (MFW) System** **A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including:** (CFR: 41.5 / 45.5) A1.03 Power level restrictions for operation of MFW pumps and valves. 2.7* 2.9*

Associated objective(s):

Byron ILT 09-1 Cert Exam

Given:

Unit 1 was at 100% power, normal alignment.

The following event occurs:

- Annunciator 1-16-E1, FW PUMP NPSH LOW, alarms due to a secondary plant transient.
- NO automatic actions associated with the alarm have occurred (assume the actuation setpoint has been reached).
- The US directs the NSO to manually perform the automatic actions associated with the alarm.

With the above conditions, the NSO will take action(s) to...

A. TRIP a running FW pump.

B. START the standby HD pump.

C. START the standby CD/CB pump.

D. OPEN 1CB025, LP HTR BYP VLV.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee ability to operate controls identified in the alarm response manual.

A low FW pump NPSH signal can be generated from various secondary transients. BAR 1-16-E1 list the auto actions of the condition. However, the auto actions do not happen until the condition is 2% below the alarm setpoint. Therefore, having the alarm come in without the actions occurring is plausible.

Choice A is incorrect, this is plausible because a similar alarm (1-16-D1, Low FW PP Suct Hdr Press) has an auto action that trips the 1A FW pump.
Choice B is incorrect, this is plausible because a probable cause of the low NPSH condition in BAR 1-16-E1 is insufficient HD pps running. However, there is no auto start feature for the HD pps.
Choice D is incorrect, this is plausible because bypassing the LP heater string is one possible method of getting more flow to the FW pp suction. It is also an auto action of a HI-2 level condition in a 11 heater.
Choice C is correct, the first auto action listed in BAR 1-16-E1 is start of standby CD/CB pump.

Ref: BAR 1-16-E1

Question 45 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28280
User-Defined ID:	2012 NRC EXAM Q#45
Cross Reference Number:	S.CD-01-06-A
Topic:	2012 NRC Exam Question #45
Num Field 1:	RO 4.2
Num Field 2:	SRO 4.0
Text Field:	059gen. 2.4.50
Comments:	Bank Question from 2011 Brwd NRC Exam Low Cog. **059 Main Feedwater (MFW) System** **2.4.50 Ability to verify system alarm setpoints and operate controls identified in the |** **alarm response manual. |** (CFR: 41.10 / 43.5 / 45.3) | IMPORTANCE RO 4.2 SRO 4.0

Associated objective(s):

DISCUSS the operation of the condensate and feedwater system with respect to plant response for the following situation: Loss of CD/CB pump

Given:

Unit 1 is at full power.

- A feedwater malfunction has resulted in NR SG water levels reaching 12%.
- Concurrently Instrument Bus 111 has been deenergized.

3 minutes later, how will 1AF005 A-D, S/G 1_ FLOW CONT VLV, respond to these conditions and in addition to locally, where can these valves be controlled from in accordance with approved plant procedures?

	1AF005A-D position	controlled from
A.	closed	Fire Hazards Panel
B.	open	Remote Shutdown Panel
C.	open	Fire Hazards Panel
D.	closed	Remote Shutdown Panel

Answer: D

Answer Explanation:

Meets the K/A.. Requires examinee knowledge of an automatic control malfunction of the Aux. Feed system and knowledge of procedural guidance should that failure occur.

On a loss of Instrument Bus 111 the "A" Train AF005 valves will fail to the closed position because the control flow signal fails to 0. If there is any flow the sensors will see it and drive the valves closed to reach setpoint. The A train AF pp starts due to an AMS signal. Per 1BEP-0 and 1BOA Elect-2 these valves can have the air failed to them, which will cause them to open and can be manually locally controlled via the associated handwheels. They could be controlled via a pneumatic signal from the remote shutdown panel if the control room were to be evacuated. When local is selected at the Remote Shutdown Panel the 3-way solenoid that supplies the actuator with air is ported to allow the air signal to be directly controlled from the RSDP (ref: 1-4030-AF05).
There are no controls at the Fire Hazards Panel for these valves but it is considered a plausible distractor because this panel can be used for remote monitoring of plant parameters, similiar to the Remote Shutdown Panel
Failing open is also considered a plausible distractor because these valves fail open on loss of air pressure.

Ref: 1BOA ELEC-2 Loss of Instrument Bus- Table A step 2.a
1-4030-AF05

Question 46 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28296
User-Defined ID:	2012 NRC EXAM RO Q46
Cross Reference Number:	T.EP01-06-A
Topic:	2012 NRC EXAM RO Q46
Num Field 1:	RO 3.1
Num Field 2:	SRO 3.4
Text Field:	061A2.05
Comments:	New question, Lo Cog **061 Auxiliary / Emergency Feedwater (AFW) System** **A2 Ability to (a) predict the impacts of the following malfunctions or operations** **on the AFW; and (b) based on those predictions, use procedures to correct,** **control, or mitigate the consequences of those malfunctions or operations:** (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.05 Automatic control malfunction 3.1* 3.4*

Associated objective(s):

GIVEN a set of plant conditions, DIAGNOSE and ANALYZE a Reactor Trip or Safety Injection

Given the following scenario:

- Your crew has entered 1BFR- H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK, UNIT 1.
- NO Auxiliary Feedwater is available.
- You have NOT met the conditions for an RCS Feed and Bleed.
- SG pressures are all between 900 and 1000 psig.

Which one of the following 6.9 KV buses would give you MAXIMUM flexibility to use the MAIN Feedwater System to restore the secondary heat sink?

Bus:

A. 156

B. 157

C. 158

D. 159

Answer: D

Answer Explanation:

Meets K/A. Requires examinee to pick the 6.9 KV bus that would provide the most flexibility to add feedwater to a SG. Indirectly, requires the examinee to assess the loss of each of the buses, thereby meeting the K/A

Bus 159 powers the Startup Feedwater Pump in addition to 2 Condensate/Condensate Booster pumps and therefore would require no further buses to supply feedwater.
Bus 158 powers 2 Condensate/Condensate Booster pumps but NO Feedwater pump.
Bus 157 powers no feedwater or Condensate/Condensate Booster pumps
Bus 156 powers the MD FW pp only.
In order to operate a feedwater pump, Condensate/Condensate Booster pump(s) must be available. BFR-H.1 specifies checking bus 159 available. 1 bus (159) would provide all pumping power required. 1BFR-H.1 later checks bus 156 available for the MDFW pump, bus this would require an additional bus available for the cond/cond. bstr pps.

Ref:1BFR -H.1 step 9d
Horsenote AC-3 AC one-line diagram

Question 47 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28297
User-Defined ID:	2012 NRC EXAM RO Q47
Cross Reference Number:	T.MI03-10
Topic:	2012 NRC Exam RO Question # 47
Num Field 1:	RO 3.5
Num Field 2:	SRO 3.9
Text Field:	062K3.01
Comments:	New Question, Lo Cog **SYSTEM: 062 AC Electrical Distribution System** **K3 Knowledge of the effect that a loss or malfunction of the ac distribution** **system will have on the following:** (CFR: 41.7 / 45.6) K3.01 Major system loads 3.5 3.9

Associated objective(s):

LIST the power supplies for equipment that could be used to maintain Critical Safety Functions during an accident

Given:

- The 1A DG has been manually started per, 1BOSR 8.1.2-1, UNIT ONE 1A DIESEL GENERATOR OPERABILITY SURVEILLANCE
- It is supplying bus 141 in PARALLEL with the associated SAT.

How will 1PM01J indications respond to an SI signal?

Indications will show.....

A. ACB 1413 to trip open, Bus 141 will remain energized from the SAT, all the ESF equipment on Bus 141 will sequence on from the Safe Shutdown Sequencer.

B. ACB 1412 to trip open, Bus 141 will be energized from the 1A DG, all the ESF equipment on Bus 141 will sequence on from the Loss of Voltage signal.

C. ACB 1413 to trip open, Bus 141 will remain energized from the SAT, all the ESF equipment on Bus 141 will start immmediately from the SI signal.

D. ACB 1412 to trip open, Bus 141 will be energized from the 1A DG and all will strip and the ESF equipment will start immediately from the SI signal.

Answer: C

Answer Explanation:

Meets K/A. Requires knowledge of Electric Plant response of safety related equipment to a safety signal actuation.

The correct answer is correct based on when an SI signal is received and both the DG and SAT are supplying the ESF Bus, the DG output breaker (1413) will trip. There will be no Loss of Voltage signal as the SAT breaker will remain closed. The Safe Shutdown sequencer will be bypassed and all associated load breakers on the ESF Bus immediately close. The DG will remain running, however it will be unloaded.

The distractors are plausible based on the students misunderstanding of what occurs in this situation.

Ref: DG L-P pg 37 and 47 of 87

Question 48 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	8
Difficulty:	0.00
System ID:	17715
User-Defined ID:	BYLC3CAP1016
Cross Reference Number:	
Topic:	BYLC3CAP1016 The 1A DG has been manually started per the mo
Num Field 1:	3.5
Num Field 2:	3.6
Text Field:	062000A3.05
Comments:	Bank Question, Hi Cog, RO level **SYSTEM: 062 AC Electrical Distribution System** **A3 Ability to monitor automatic operation of the ac distribution system,** **including:** (CFR: 41.7 / 45.5) A3.05 Safety-related indicators and controls . . . . 3.5 3.6 Reference: 6E-1-4030AP, 6E-1-4030EF, BOSR 8.1.2-1, BOP DG-11

Associated objective(s):

ANALYZE a set of plant conditions and DETERMINE the proper response of the 6.9kv and 4kv busses, including overall plant response for the following:

Old Import of questions - Review for deletion

Given:
- Unit 1 has just tripped from 100% power concurrent with a Loss of DC Bus 112.
- All equipment operates as designed.
- During performance of recovery procedures, the NSO pushes the MCR "trip" pushbuttons for BOTH Unit 1 Turbine Driven FW Pumps.

With NO other operator actions, the status of the 1B and 1C FW pumps is…

	1B FW Pump	1C FW Pump
A.	tripped	tripped
B.	running	tripped
C.	running	running
D.	tripped	running

Answer: B

Answer Explanation:

Question meets KA - question requires examinee knowledge of bus power supplies to 125 VDC loads.
With a loss of DC bus 112, DC bus 114 is also lost. DC bus 114 is the control power to the 1B FW pump, therefore the MCR trip push button will not work for 1B FW pp. 1C FW pp is supplied by DC bus 113, so it will trip when P/B is depressed.
A is correct, see above explanation
B is incorrect, see above explanation
C is incorrect, see above explanation
D is incorrect, see above explanation
All distractors are plausible based on students misunderstanding of the FW pp and associated DC power supplies.

Ref: BOA ELEC-1 attach. C step 3 (pg. 25)

Question 49 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28259
User-Defined ID:	2012 NRC EXAM RO Q49
Cross Reference Number:	
Topic:	2012 NRC Exam RO Question 49
Num Field 1:	RO 2.9
Num Field 2:	SRO 3.1
Text Field:	063K2.01
Comments:	Bank Question from Bwd 2011 Cert. Exam Low Cog RO Level **063 D.C. Electrical Distribution** **K2 Knowledge of bus power supplies to the following:** (CFR: 41.7) K2.01 Major DC loads 2.9* 3.1

Associated objective(s):

PREDICT how supported systems will be impacted by various 125 VDC System failures

Unit 2 is operating at 90% power with all control systems in automatic.

- Annunciator 2-22-E7, 125V DC BATT CHGR 212 FD BRKR TRIP, has alarmed.

The following indications are provided:
- DC Bus 212 voltage indicates 129 VDC
- Annunciator 2-22-E10, 125V DC PNL 212/214 VOLT LOW, is NOT Lit.

What is the current status of DC Bus 212/214 and indicate whether or NOT Tech Specs have to be entered?

A. DC Busses 212 and 214 are energized by Battery Charger 212. Tech Specs do NOT have to be entered.

B. DC Busses 212 AND 214 are energized by DC Battery 212. Tech Specs have to be entered.

C. DC Bus 212 is energized by Battery 212 but DC Bus 214 is de-energized. Tech Specs do NOT have to be entered.

D. DC Busses 212 and 214 are energized and have low Voltage. Tech Specs have to be entered.

Answer: B

Answer Explanation:

Meets K/A. Requires examinee to assess Control Board meters and annunciators to assess the status of the DC Electrical Distribution System.

When the Battery Charger A.C. feed breaker trips, DC voltage will not be lost to the 212 and 214 DC Busses. The charger is lost but the battery will supply voltage for DC loads until an appropriate power source can be obtained. This would be either X-tie to Unit-1 DC system or to restore the dedicated charger to operable status. We will enter Tech. Spec. 3.8.4 under this condition. All selections are plausible based on students knowledge of the DC system, alarms, indications and Technical Specifications.

Ref: BAR 2-22-E7

Question 50 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28298
User-Defined ID:	2012 NRC EXAM RO Q50
Cross Reference Number:	T.OA01-02
Topic:	2012 NRC Exam RO Question #50
Num Field 1:	RO 2.7
Num Field 2:	SRO 3.1
Text Field:	063A3.01
Comments:	New Question- High Cog **063 D.C. Electrical Distribution** **A3 Ability to monitor automatic operation of the DC electrical system,** **including:** (CFR: 41.7 / 45.5) A3.01 Meters, annunciators, dials, recorders, and indicating lights 2.7 3.1

Associated objective(s):

ANALYZE a given set of plant conditions and DETERMINE if entry into 1/2BOA ELEC-1, Loss of DC Bus, is required

Unit 1 is at 100% power and in a normal full power alignment.

- A Loss of off-site power occurs due to a System Auxiliary Transformer fault.
- The fault is subsequently CLEARED by automatic breaker operation.
- Both Unit 1 Emergency Diesel Generators start and load as designed EXCEPT:

- Half way through the sequencing of the 1A DG, a governor problem occurs that will NOT allow the fuel racks to open any further.

Under these conditions, which of the following listed Breaker OR Engine trips is most likely to occur as the 1A DG continues to load?

A. Neutral Overcurrent

B. Reverse Power

C. Engine Overspeed

D. Under Frequency

Answer: D

Answer Explanation:

Meets K/A. Requires examinee knowledge of trips that are enabled during a loss of voltage event and the affects of a governor malfunction during loading of the diesel generator.

If the fuel rack is "frozen" in the current position and load on the machine is increased as postulated by the question, the machine will slow down. This results in lowering frequency. From the selection of electrical trips, the underfrequency is the only one that is correct under theses conditions.
Neutral overcurrent is plausible, however only occurs during a fault condition between the source (diesel generator) and a load. No fault is postulated.
Reverse power is considered plausible however is not credible because the generator in this case is still supplying current rather than acting like a load.
Engine overspeed is plausible if the examinee thinks the EDG will shed it's load, and with frozen fuel racks won't slow down

Ref: Horse Note DG-1
DG L-P pg 36 and 37

Question 51 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28299
User-Defined ID:	2012 NRC EXAM RO Q51
Cross Reference Number:	3C.DG-02-A
Topic:	2012 NRC EXAM RO Question #51
Num Field 1:	RO 3.8
Num Field 2:	SRO 4.1
Text Field:	064K4.1
Comments:	New Question, High Cog **064 Emergency Diesel Generator (ED/G) System** **K4 Knowledge of ED/G system design feature(s) and/or interlock(** **s) which provide for the following:** (CFR: 41.7) K4.01 Trips while loading the ED/G (frequency, voltage, speed) 3.8 4.1

Associated objective(s):

DESCRIBE plant parameters and system response to the loading of a Diesel Generator.

Given:

You are in the process of performing a Gaseous Release of 0A GDT.

Concurrently, a High Rad bag of material must be moved past 0PR02J, GAS DECAY TANK EFFLUENT RADIATION MONITOR.

Which of the below cases will result in the GREATEST probability of automatically terminating the release due to moving the High-Rad. bag past the rad monitor?

? Case 1: walking 1 meter away from 0PR02J with a 5 second transit time.
? Case 2: walking 2 meters away from 0PR02J with a 20 second transit time.
? Case 3: walking 3 meters away from 0PR02J with a 1 minute transit time.
? Case 4: walking 5 meters away from 0PR02J with a 5 minute transit time.

Consider the bag as a point source.

The release is MOST likely to be automatically terminated during Case...

A. 1

B. 2

C. 3

D. 4

Answer: A

Answer Explanation:

Meets KA. Requires examinee to assess that radiation dose rate changes as distance changes and tests if the examinee knows what the rad. monitor is really looking at.

The correct answer has nothing to do with time, but only with distance. The closest distance for a given source strength will yield the greatest dose rate.
All distractors are plausible if the examinee confuses dose with dose rate.

Ref: NanTel Nuclear General Employee Study Guide- page 102 dose and dose rate, pg 113 reducing dose-distance.

Question 52 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28344
User-Defined ID:	2012 NRC EXAM RO Q52
Cross Reference Number:	S.AR1-04-B-1
Topic:	2012 NRC Exam RO Question 52
Num Field 1:	RO 2.5
Num Field 2:	SRO 3.1
Text Field:	073K5.02
Comments:	New Question, High Cog, RO level **073 Process Radiation Monitoring (PRM) System** **K5 Knowledge of the operational implications as they apply to** **concepts as they apply to the PRM system:** (CFR: 41.5 / 45.7) K5.02 Radiation intensity changes with source distance 2.5 3.1

Associated objective(s):

STATE the interlocks associated with the AR/PR system and purpose of each including:
PR: Liquid Radwaste Effluent

The Essential Service Water System can supply a BACK-UP water supply to which of the following systems?

A. Main Feed Water

B. Spent Fuel Pool

C. Demineralized Water

D. Fire Protection

Answer: D

Answer Explanation:

Meets K/A. Tests examinee on the physical relationship between SX and Fire Protection system

Fire Protection is the correct answer
Main Feed Water is plausible because SX can provide for Auxiliary Feedwater
Spent Fuel Pool is cooled by Component Cooling water but since CC is cooled by SX it may discriminate the examinees.
Deminerailzed Water is a plausible distractor based on a proposed modification that will allow SX as an alternate make-up supply to the Component Cooling Water system

Ref: SX Lesson plan pg 27 and 35

Question 53 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28301
User-Defined ID:	2012 NRC EXAM RO Q53
Cross Reference Number:	S.SX1-03
Topic:	2012 NRC Exam RO Question #53
Num Field 1:	RO 2.5
Num Field 2:	SRO 2.6
Text Field:	076K1.15
Comments:	New Question, Low Cog **076 Service Water System (SWS)** **K1 Knowledge of the physical connections and/or cause-effect relationships** **between the SWS and the following systems:** (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.15 FPS 2.5 2.6

Associated objective(s):

LIST the load cooled by Essential Service Water

Given:

- Unit 1 is at 50% power in a normal line-up for this power level.
- Unit 2 is at 28% power and "holding" after a refueling outage.
- 2 Feedwater pumps are running on Unit 2.
- NO unplanned LCOARs are in effect on EITHER unit.
- Instrument air is cross-tied between units.

A large loss of IA occurs rapidly lowering Unit 1 and Unit 2 IA pressure to 15 psig.

With NO operator action, what will be the status of EACH Reactor unit, 20 minutes following the LOSS of Instrument Air?

	Unit 1 Reactor	Unit 2 Reactor	Reason
A.	Tripped	Operating	Unit 1: High PZR water level. Unit 2: 20 minutes is NOT enough time to reach Lo-2 SG level setpoint.
B.	Operating	Operating	Unit 1: 20 minutes is NOT enough time to reach Lo-2 SG level setpoint. Unit 2: is operating because Rx power is < P-8.
C.	Tripped	Tripped	Unit 1: on Lo-2 SG water level. Unit 2: on Lo-2 SG water level.
D.	Tripped	Tripped	Unit 1: on Lo-2 S/G level. Unit 2: on Hi PZR level.

Answer: C

Answer Explanation:

Meets K/A. Requires knowledge of effects, as a result of loss of I.A. on both units when crosstied.

FWRVs fail closed on loss of IA. This results in lowering SG water level until the trip setpoint is reached. On unit 1 this will happen within about 6 minutes from the inception of the loss, as data shows from the simulator. Unit 2 will take a longer period of time (approximately 2 times longer) but will happen within the 20 minutes specified.
All other selections are plausible as the examinee will have to differentiate which trips are active below P-8.
Letdown is lost on a loss of IA which will result in a level rise in the pressurizer, making it plausible. The High PZR level trip is at 92% and will take > 20 minutes to reach at minimum charging flow and 128 gallon per percent PZR level.

Ref: 1BOA SEC-4 Table A pg 1 of 7 states FWRVs fail closed which will result in a Rx Trip w/no operator action.

Question 54 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28302
User-Defined ID:	2012 NRC EXAM RO Q54
Cross Reference Number:	3D.OA-25-A
Topic:	2012 NRC exam RO Question 54
Num Field 1:	RO 3.0
Num Field 2:	SRO 3.4
Text Field:	078K3.03
Comments:	New Question, High Cog **078 Instrument Air System (IAS)** **K3 Knowledge of the effect that a loss or malfunction of the IAS will have on the following:** (CFR: 41.7 / 45.6) K3.03 Cross-tied units 3.0 3.4

Associated objective(s):

DISCUSS the response of integrated plant and system critical parameters to a Loss of Instrument Air, with and without operator action

Given:
- Unit 2 is in Mode 1.
- The 2A, 2B, and 2D RCFC's are operating in high speed.
- The 2C RCFC is in standby.

The following indications are observed on the Unit 2 RCFC Dry Bulb temperatures:
- 2A RCFC Inlet Temperature 119° F.
- 2B RCFC Inlet Temperature 118° F.
- 2C RCFC Inlet Temperature 127° F.
- 2D RCFC Inlet Temperature 121° F.

Per the Tech Spec 3.6.5, Containment Air Temperature...

A. NO action is necessary because ALL the individual RCFC temperatures are within the appropriate LCO limit(s).

B. the action requirement must be applied because ONE of the OPERATING RCFC's temperatures is above the LCO upper limit.

C. the action requirement must be applied because the average of ALL the RCFC temperatures exceeds the LCO upper limit

D. NO action is necessary because the average temperature of ALL OPERATING RCFC's is below the LCO upper limit.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee ability to monitor system parameters to prevent exceeding design limits (Technical Specifications).
1BOSR 0.1-1,2,3, Modes 1,2,3 Shiftly Daily Operating Surv. step F.8 (pg. 10) describes the method for calculating containment temperature for tech spec limit comparison. This method is to calculate the average of the inlet temperatures on the running RCFCs. (pg. 38)
Choice A is incorrect because the shut down RCFC is not calculated into the average.
Choice B is incorrect because only one RCFC over the limit does not require TS entry.
Choice C is correct, see explanation above.
Choice D is incorrect because 2D RCFC is above the limit

Ref: T/S 3.6.5, not included with submittal package.

Question 55 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28262
User-Defined ID:	2012 NRC EXAM RO Q55
Cross Reference Number:	3C.VP-06-C
Topic:	2012 NRC Exam RO Question 55
Num Field 1:	RO 3.7
Num Field 2:	SRO 4.1
Text Field:	103A1.01
Comments:	From 2011 Brwd NRC Exam High Cog RO Level **103 Containment System** **A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including:** (CFR: 41.5 / 45.5) A1.01 Containment pressure, temperature, and humidity 3.7 4.1

Associated objective(s):

Given a set of plant conditions or parameters involving RCFCs, DETERMINE if entry into a Tech Spec action statement is required

During a Unit One Reactor Startup, from a reactor trip 5 days ago and at 6400 EFPH, the Reactor Operator logged the following information when withdrawing Control Banks:

NOTE: Prior to Shutdown Bank withdrawal, source range counts were 30 CPM.

Control Bank Position	Source Range Count Rate in CPM
• Bank A	
0 steps	70
50	100
113	180
• Bank B	
50	300
113	430
• Bank C	
25	560

Which ONE of the following describes the condition of the reactor and proper operator action for the above conditions? (BCB-1 Figure 9 is attached)

A. The Reactor IS critical below the LO-2 Rod Insertion Limit and must be tripped.

B. The reactor is NOT critical but criticality is predicted below the Lo-2 Rod Insertion Limt and boration is required.

C. The Reactor is NOT critical and a normal startup may continue.

D. The Reactor IS critical and immediately perform a Shutdown Margin Calculation.

Answer: B

Answer Explanation:

Meets K/A. Requires the examinee's knowledge of the relationship of the 8-fold count and criticality. Requires the examinee anticipation of criticality during positive reactivity additions during a reactor startup evolution. Requires examinees knowledge of the Shutdown Margin Tech. Specs, in addition to using a Figure from the Byron Curve Book. The examinee must also be familiar with the Reactor Startup procedure, 1BGP 100-2A1.

The correct answer is correct based on the correct reading of the reference and the realization that Mode 2 SDM is not satisfied and that the procedure and Tech. Spec. require boration to restore SDM.

The distractors indicating the reactor is critical are plausible if the examinee thinks the 8 fold count is related prior to any rod withdrawl.

The distractor that states the reactor is not critical and a normal startup may continue is considered plausible due to errors reading the graph.

Ref. BCB-1 Fig. 9 with burnup at 6452.8 EFPH
BGP 100-2A1 step F.7.m (pg. 11) and Att. A (pg 16)

Question 56 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28303
User-Defined ID:	2012 NRC EXAM RO Q56
Cross Reference Number:	T.GP02-06
Topic:	2012 NRC Exam RO Question 56
Num Field 1:	RO 4.2
Num Field 2:	SRO 4.3
Text Field:	001K5.18
Comments:	Bank Question High Cog modified distractors of question wfh 8/9/11 **001 Control Rod Drive System** **K5 Knowledge of the following operational implications as they** **apply to the CRDS:** (CFR: 41.5/45.7) K5.18 Anticipation of criticality at any time when adding positive reactivity during startup 4.2 4.3

Associated objective(s):

ANALYZE a given set of plant conditions and DETERMINE the required actions per GP 100-2, Plant Startup

Given:

- A Loss of all AC power has occurred on Unit 1.
- The crew has implemented 1BCA 0.0, LOSS OF ALL AC POWER-UNIT 1.

- The crew has performed Step 14, ENERGIZE BUS 141 USING LIMITED UNIT 2 CROSSTIE.
- The crew is currently performing Step 23, RESTORE UNIT 1 CENT CHG PUMP.

What will power the 1A CV pump using this method?

A. SAT Transformer 242-1.

B. 2A Diesel Generator.

C. UAT Transformer 141-1.

D. SAT Transformer 242-2.

Answer: B

Answer Explanation:

Meets K/A by requiring examinee to know the power supply of the 1A CV pump during a limited cross-tie evolution while implementing 1BCA 0.0.

The correct answer is from the 2A DG, that is the limited cross-tie portion.
SAT 242-1 is plausible but not correct because it is not "limited" cross-tie.
SAT 242-2 is plausible but is not the correct train.
Supplied by U-1 UAT is plausible based on examinee's misconceptions

Ref: 1BCA 0.0 step 14 (pg. 19)

Question 57 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28304
User-Defined ID:	2012 NRC EXAM RO Q57
Cross Reference Number:	3D.CA-01-E
Topic:	2012 NRC exam RO question 57
Num Field 1:	RO 3.1
Num Field 2:	SRO 3.2
Text Field:	011K2.01
Comments:	New question, High cog., RO level **011 Pressurizer Level Control System** **K2 Knowledge of bus power supplies to the following:** (CFR: 41.7) K2.01 Charging pumps 3.1 3.2

Associated objective(s):

Given a set of plant conditions or parameters indicating a Loss of All AC and a set of plant procedures, IDENTIFY the correct procedure(s) to be utilized and DISCUSS required operator actions

Given the following conditions on Unit 1:

? The reactor is at FULL power.
? PR channel N-43 failed HIGH last shift and ALL actions of 1BOA INST-1 "NI MALFUNCTION" have been completed.
? Instrument Bus 114 has just lost power.

After the reactor trip, what will be the status of the P-7, and P-10, status lights on the BYPASS PERMISSIVE PANEL?

	P-7	P-10
A.	Lit	NOT LIT
B.	NOT Lit	NOT Lit
C.	NOT Lit	Lit
D.	Lit	Lit

Answer: B

Answer Explanation:

Meets K/A requires the knowledge to interpret the NI inputs to the Reactor Protection System and predict how a loss of power will affect system response.

P-10 will not be lit. In order for it to light 3 of 4 power range channels must be less than 10% NI power. The combination or tripped bi-stables and the loss of power results in both channel (N43 and 44) failing high. They will not input in the current state to make up the 3 of 4 inputs to P-10. This results in P-10 not being lit. For P-7 to light, BOTH P-13 AND P-10 must be enabled. P13 will be made up but again P-10 will not. Therefore P-7 will not be made up and will remain dark or NOT Lit.

All distractors are plausible based on examinee's knowledge of the inputs to these permissives.

Ref:
BAR 1-BP-3.4
BAR 1-BP-3.5

Question 58 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28305
User-Defined ID:	2012 NRC EXAM RO Q58
Cross Reference Number:	3C.NI-04-D
Topic:	2012 NRC Exam RO question 58
Num Field 1:	RO 4.1
Num Field 2:	SRO 4.2
Text Field:	015K1.01
Comments:	New question, High Cog **015 Nuclear Instrumentation System** **K1 Knowledge of the physical connections and/or causeeffect relationships between the NIS and the following systems:** (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.01 RPS 4.1 4.2

Associated objective(s):

DESCRIBE the interlocks and functions associated with the Power Range Detectors

When a Core Exit Thermocouple fails LOW, the average of the 10 highest CETC ...
(assume NO operator action)

A. LED display will flash.

B. LED display will go DARK.

C. SPDS subcooling display will flash.

D. LED display will indicate a LOW number.

Answer: A

Answer Explanation:

Meets K/A. Requires examinee knowledge of the operation and display functions of the CETC system. Also requires knowledge of CETC failures and what will be displayed based on those failures. Note that actions to prevent exceeding design limits are excluded from this question to concentrate on the monitoring changes in parameters portion of the K/A.

The correct answer is correct based on reference BOP RC-12.
When a CETC fails low, it will no longer be considered one of the 10 highest CETC inputs into the Subcooled Margin Monitor but is considered a plausible distractor.
The display going dark is considered a plausible distractor.
Read low is considered plausible if the examinee does not know that it will be no longer included into the "10 highest" calc.

Ref: BOP RC-12, step 7 (pg. 4)

Question 59 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28345
User-Defined ID:	2012 NRC EXAM RO Q59
Cross Reference Number:	S.IT1-05
Topic:	2012 NRC Exam RO Question 59
Num Field 1:	RO 3.7
Num Field 2:	SRO 3.9
Text Field:	017A1.01
Comments:	New Question, Low Cog., RO Level **017 In-Core Temperature Monitor (ITM) System** **A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ITM system controls including:** (CFR: 41.5 / 45.7) A1.01 Core exit temperature 3.7 3.9

Associated objective(s):

EXPLAIN the principles of operation for the Core Exit Thermocouple System

Which of the following is a direct entry condition for 0BOA Refuel-3, Loss of Spent Fuel Pit Cooling Unit 0?

A. Spent Fuel Pit Heat Exchanger outlet temperature of 125 degrees F.

B. Annunciator 1-1-C1, SPENT FUEL PIT LOW LEVEL, in alarm.

C. Annunciator 1-1-A1, SPENT FUEL PIT PUMP TRIP, in alarm.

D. Area Rad monitor 0AR37J, FUEL HANDLING BUILDING GENERAL AREA, in ALERT.

Answer: C

Answer Explanation:

Meets K/A. Requires knowledge that Annunciator 1-1-A1 may require entry into 0BOA Refuel-3.

Correct answer is correct as stated in both the annunciator response and 0BOA Refuel-3.
A SFP HX temperature is plausible but the alarm and entry condition to 0BOA Refuel-3 is at 149 degrees not 125.
A spent fuel pool low level alarm is again plausible but not applicable to this condition. The correct BOA would be Unit specific Refuel 2, Refuel cavity or SFP Level Loss
High general area radiation is considered plausible because Refuel-3 addresses informing RP of condition but again is not an entry condition. It is an entry condition to 1/2 BOA Refuel 1 and 2.

Ref: BAR 1-1-A1 and 0BOA Refuel -3, Loss of Spent Fuel Cooling, (pg. 1)

Question 60 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	17890
User-Defined ID:	BWLI-REF3001
Cross Reference Number:	T.OA31-02
Topic:	BWLI-REF3001
Num Field 1:	RO 4.2
Num Field 2:	SRO 4.1
Text Field:	033G2.4.31
Comments:	Bank Question, Low Cog **033 Spent Fuel Pool Cooling System (SFPCS)** 2.4.31 **Knowledge of annunciator alarms, indications, or response procedures.** | (CFR: 41.10 / 45.3) | IMPORTANCE RO 4.2 SRO 4.1

Associated objective(s):

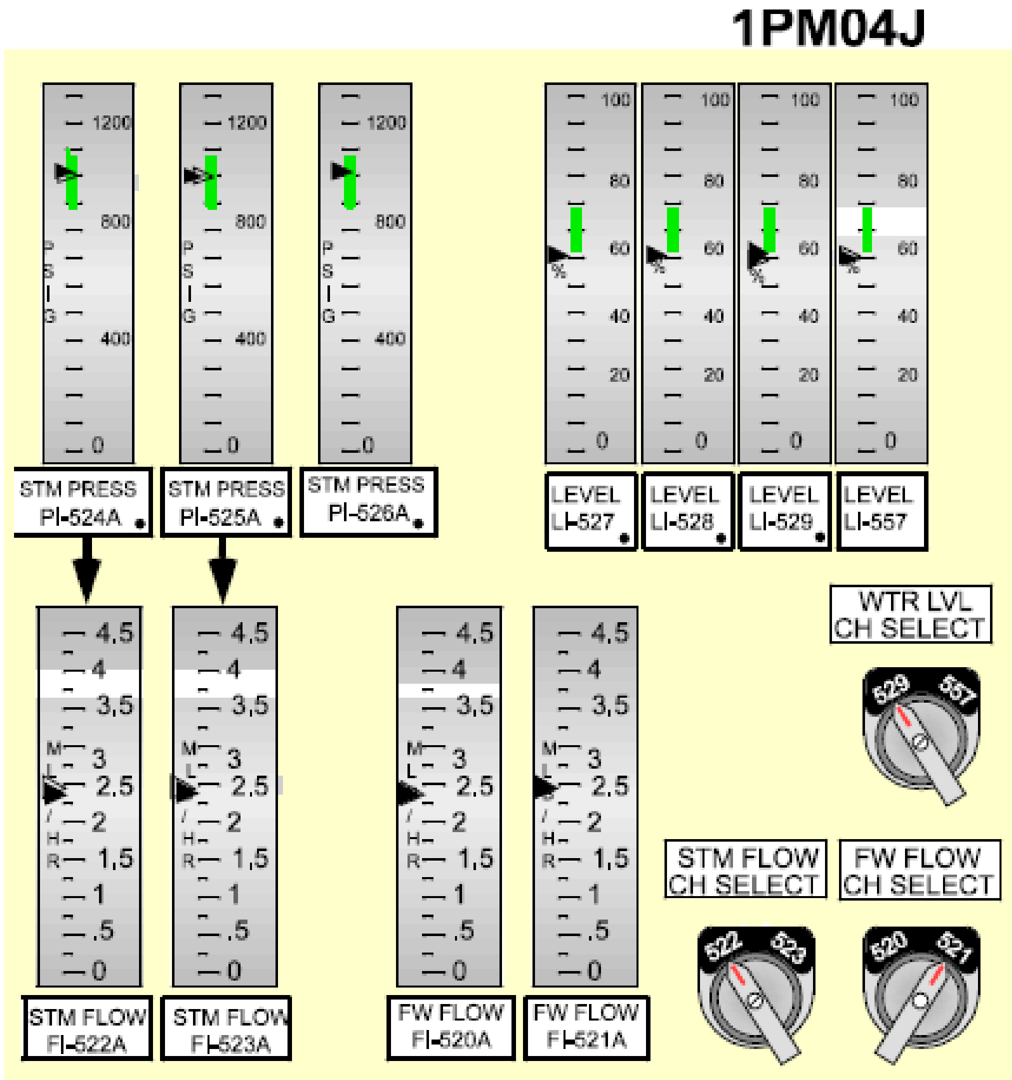
ANALYZE a given set of plant conditions and DETERMINE if entry into 0BOA REFUEL-3, Loss of Spent Fuel Pit Cooling, is required

Unit 1 is at power with the following conditions:

- 1B and 1C Main Feed Pumps are online
- 1PT-524, Steam Pressure fails high

NOTE: The indications depicted below are JUST PRIOR to the event.

Select the statement that describes the response with **NO** operator actions. (Assume shrink and swell are negligible for these conditions.)



A. All SG levels drop initially and MFPs speed goes up.

B. 1B FRV opens initially and MFPs speed goes down.

C. 1B SG level rises initially and MFPs speed goes up.

D. All FRVs close initially and MFPs speed goes down.

Answer: C

Answer Explanation:

Meets K/A. Requires examinee to monitor the automatic operation of the SG water level control system

Density compensation is a multiplier for steam flow. Density goes down as steam pressure drops. At low power, higher pressure is a larger multiplier. So when the transmitter fails high, the multiplier is the largest. At this power it will bring the indicated steam flow up from about 2.4 Mlb/hr to 2.8 Mlb/hr. The sum of the steam flows feed the DP Program of % Full Power Steam vs. Program DP. The Master Controller looks at Actual (PT-507, Stm Hdr and PT-508, FW Pump Disch Hdr). It sees the actual lower than the program and DP program calls for a higher DP, thus MFPs speed will go up.

The FRV opens, due to Steam flow greater than feedwater flow, a rapid response. The FRV will open until the level signal SG level rises, long term response and back down on the Feedwater reg. valve, in about 1 minute. The Steam generator level will settle out at a slightly higher value.

All other response are plausible based on the examinee's mis-understanding of the SGWLC system

Ref: SGWLCS L-P pages 19 (Steam press. compensation channel failure), 21 (Steam Flow channel failure) and pg 26, (Operation of 3 element controller)
and FW pp Speed Control pg. 73 (question 12 on practice exercise)

Question 61 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	20752
User-Defined ID:	BYLC3CSG03C001
Cross Reference Number:	3C.SG-03-C
Topic:	BYLC3CSG03C001 - 50% Both MFPs on and PT-514, Steam Pressure fails high
Num Field 1:	RO 4.0
Num Field 2:	SRO 3.9
Text Field:	035A3.01
Comments:	Bank Question High Cog. **035 Steam Generator System (S/GS)** **A3 Ability to monitor automatic operation of the S/G including:** (CFR: 41.7 / 45.5) A3.01 S/G water level control 4.0 3.9

Associated objective(s):

Given a set of plant conditions or parameters, EXPLAIN how the SGWLC System functions to maintain program level

Given:

A Reactor trip has just occurred on Unit 1 from 75% power. The following plant conditions exist after the Reactor trip:

- Train A Reactor Trip Breaker: Open
- Train B Reactor Trip Breaker: Closed
- DRPI Rod Bottom lights: all LIT
- Neutron flux: Lowering

The setpoint at which Tave will be controlled by the Steam Dumps is ____ degrees F.

A. 550

B. 557

C. 560

D. 561

Answer: C

Answer Explanation:

Meets K/A. Tests examinee on knowledge of RCS temperature control by the Steam Dump system following a Reactor Trip due to a failure of the Reactor Trip Controller, that resulted from RTB B failing to open. Normal SD response is recall but diagnosis of a failure is comprehension.

The steam dumps will be controlled by the Load Rejection controller. At 0% load a 3 degree offset is programmed in from a no-load Tave condition. (i.e. 560 degrees vs. the normal 557). Since RTB didn't open, the steam dump system will not know the reactor tripped to maintain 557. The steam dumps will respond to a 560 degree setpoint.
550 degrees is a plausible distractor as that is the temperature at which the steam dumps are no longer "armed" . If the examinee would mistakenly think the dumps would stay open until P-12 is met, this is where they would control at.
561 is a plausible distractor because if the steam dumps do not open RCS temperature would be controlled by the SG PORVs at 561 degrees.

Ref: Steam Dump Lesson plan pg 23.

Question 62 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28307
User-Defined ID:	2012 NRC EXAM RO Q62
Cross Reference Number:	3C.DU-01-C
Topic:	2012 NRC Exam RO Question 62
Num Field 1:	RO 2.7
Num Field 2:	SRO 2.9
Text Field:	041K6.03
Comments:	Bank Question, High Cog, RO level **041 Steam Dump System (SDS)/Turbine Bypass Control** **K6 Knowledge of the effect of a loss or malfunction on the following will have** **on the SDS:** (CFR: 41.7 / 45.7) K6.03 Controller and positioners, including ICS, S/G, CRDS 2.7 2.9

Associated objective(s):

ANALYZE a set of plant conditions involving the Steam Dump System pertaining to the RCS temperature control and DETERMINE the proper operator action.

Answer the following two (2) questions concerning the Waste Gas Decay Tanks.

- The in-service waste gas decay tank will auto isolate at __(1)__ psig.
- The waste gas decay tanks are protected from overpressure by a relief valve that lifts at __(2)__ psig.

	(1)	(2)
A.	85	150
B.	85	100
C.	95	150
D.	95	100

Answer: C

Answer Explanation:

Meets K/A. Requires examinee to have knowledge of design features and interlocks associated with pressure of the Waste Gas Decay Tanks.

The correct answer is correct based on Lesson Plan information.
All other distractors are plausible based on examinee's knowledge and the PRT has a 100 # relief.

Ref: Radioactive Waste Gas L-P. 95# setpoint is found on pg 24
Relief valve setpoint is found on pg. 9

Question 63 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28346
User-Defined ID:	2012 NRC EXAM RO Q63
Cross Reference Number:	3C.GW-01-B
Topic:	2012 NRC Exam RO Question 63
Num Field 1:	RO 2.6
Num Field 2:	SRO 3.0
Text Field:	071K4.01
Comments:	New Question, Low Cog., RO level **071 Waste Gas Disposal System (WGDS)** **K4 Knowledge of design feature(s) and/or interlock(s) which provide for the following:** (CFR: 41.7) K4.01 Pressure capability of the waste gas decay tank 2.6 3.0

Associated objective(s):

Given the appropriate procedure and plant parameters, DESCRIBE the "Operating Section" for a Gas Decay Tank Release

While preparing to use the RM-11 console to perform a surveillance, you notice the key board is in the SUPERVISOR MODE. The concern is...

A. incorrect manipulations could change alarm setpoints for safety related monitors ONLY.

B. data retrieval is impossible since the key board is not in NORMAL.

C. ONLY Tech. Spec. monitors will continue to monitor.

D. incorrect manipulations could change alarm setpoints for non-safety related monitors ONLY.

Answer: D

Answer Explanation:

Meets K/A requires knowledge of manual operation of the RM-11 in the Control room and the effects that will have on the Area Radiation Monitors.

Per BOP AR PR-5 the answer is correct.
The alarm setpoints cannot be affected for Safety Related monitors at the RM-11. The RM-23 must be utilized.
Disabling the data retrieval function is considered a plausible distractor.

Ref: BOP AR/PR-5 E (Limitations and Actions) 1.b (pg. 2)

Question 64 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	2
Difficulty:	0.00
System ID:	17146
User-Defined ID:	BYLC3CAR3001
Cross Reference Number:	3C.AR-03-A
Topic:	BYLC3CAR3001 While preparing to use the RM-11 console to pe
Num Field 1:	2.5
Num Field 2:	2.5
Text Field:	072A4.02
Comments:	Bank Question, Low Cog **072 Area Radiation Monitoring (ARM) System** **A4 Ability to manually operate and/or monitor in the control room:** (CFR: 41.7 / 45.5 to 45.8) A4.02 Major components 2.5* 2.5

Associated objective(s):

Given the appropriate procedure, DESCRIBE how to operate the General Atomics Radiation Monitoring System.

Old Import of questions - Review for deletion

Given:

- Both units are at 100% power.
- The following equipment is OOS.
  - 1B CW Pump
  - 0A WS Pump
  - 1B SAC
- All other equipment is normally aligned.

The following occurs:
- Unit 1 reactor is manually tripped.
- All equipment operates as designed.

ONE MINUTE LATER, bus 143 faults and is deenergized.

With the above conditions and NO operator actions, which of the following will occur?

A. Unit 1 Pzr Backup Heater Group A energizes.

B. Unit 1 loses Instrument Air.

C. WS header pressure drops on BOTH units.

D. ALL unit 1 steam dump valves close.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of loss of CW system will have on ESFAS (C-9 interlock).

Choice A is incorrect, B/U heater group A is powered from bus 143, so the group will not energize. The distractor is credible because if the non-ESF bus loss was 144, then the variable heater group on U-1 would loss power and the backup heaters on the energized bus would energize on low Prz pressure.
Choice D is correct, A bus 143 fault will cause a loss of bus 143 and the 1A and 1C CW pumps brkrs opening on UV. With the 1B CW pp previously OOS, C-9 interlock (ALL CW pp brks open) would prevent steam dumps from arming and cause all steam dump valves to close.
Choice B is incorrect, 1B SAC (which is OOS) is powered from bus 144, The 1A SAC will be lost on a loss of Bus 143 but Service Air supply will not be lost due to cross-connection between units. The loss of bus 143 will not affect the status of instrument air on Unit 1.
Choice C is incorrect, 0A WS pump (which is OOS) is powered from bus 143, so the loss of bus 143 will not affect the status of WS.

Ref: BAR 1-BP-5.6

Question 65 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28272
User-Defined ID:	2012 NRC EXAM RO Q65
Cross Reference Number:	3C.EF-01-B
Topic:	2012 NRC Exam RO Question # 65
Num Field 1:	RO 3.4
Num Field 2:	SRO 3.5
Text Field:	075K3.07
Comments:	From 2009 Braidwood NRC Exam High Cog. RO Level **075 Circulating Water System** **K3 Knowledge of the effect that a loss or malfunctions of the circulating water system will have on the following:** (CFR: 41.7 / 45.6) K3.07 ESFAS 3.4* 3.5

Associated objective(s):

Given a set of plant conditions or parameters and ESFAS indications DETERMINE if the ESFAS is responding correctly.

Given:

- Unit 2 is operating at full power.
- 3 minutes ago a feedwater malfunction occured that resulted in reactor power RISING.
- The unit is still responding to the effects of the transient.
- Currently the 1 minute calorimetric indicates 101.4% power and is RISING very slightly.
- The 10 minute calorimetric is currently exceeding 100% power.

Which of the following would result in the GREATEST turbine output while restoring thermal output to within limits per the Unit Operating License?

A. Immediately initiate a Feedwater Runback for 1 minute.

B. Immediately initiate a Heater Drain Runback for 1 minute.

C. Initiate a Heater Drain Runback for 1 minute after the transient subsides.

D. Initiate a Feedwater Runback for 1 minute after the transient subsides.

Answer: B

Answer Explanation:

Meets K/A. Question requires examinee to make a conservative decision I.A.W. 2BGP 100-3. Paragraph provided below as explanation.

Unplanned activities that result in an increase in reactor power and cause the 10 minute calorimetric to exceed 100% SHALL be immediately addressed to limit the time that the 10 minute calorimetric exceeds 100%. Do NOT wait for transient conditions to subside, take prompt conservative action to reduce power to less than or equal to 100%.
The heater drain runback has a 20 MW/min ramp rate. At the end of 1 minute power will therefore be reduced by about 2%.
The FW runback rate is 250 MW/min. Therefore after 1 minute, power will be down to about 80%.
As stated above, the Reactor Operator should not wait for the transient to subside before taking action.

Ref: 2BGP 100-3 E.1.i. pg 12

Question 66 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28309
User-Defined ID:	2012 NRC EXAM RO Q66
Cross Reference Number:	3E.AM-031-K
Topic:	2012 NRC Exam RO Question 66
Num Field 1:	R.O 3.6
Num Field 2:	SRO 4.3
Text Field:	Gen. 2.1.39
Comments:	New Question, High Cog, RO level **2.1 Conduct of Operations** 2.1.39 **Knowledge of conservative decision making practices.** | (CFR: 41.10 / 43.5 / 45.12) IMPORTANCE RO 3.6 SRO 4.3

Associated objective(s):

ANALYZE a set of conditions pertaining to an Operating Policy Statement and DETERMINE the necessary operator action to comply with the policy

Unit 1 was at full power when an earthquake occurred.

- A Reactor Trip first out annunciator is LIT.
- The reactor has NOT Tripped.
- The turbine has NOT Tripped.
- The Turbine is being run back manually.
- Control Rods are being inserted into the core.
- The first group of steam dumps are OPEN.
- The crew has entered 1BFR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS UNIT 1.

Given the above situation, which of the following will give the BEST indication of core power?

A. Loop Delta T

B. Calorimetric

C. Turbine first stage pressure

D. Tave

Answer: A

Answer Explanation:

Meets K/A. Requires examinee to evaluate each selection to provide the most reliable indication of core power when the majority of the NI's are considered suspect.

Loop delta T will give the most reliable indication of core power since power is proportional to dT.
Turbine first stage pressure will not be as reliable because turbine power could be different than reactor power during the transient as the steam dumps may be open and therefore turbine power is not an accurate measure of steam flow.
Calorimetric power will not accurately track the rapid power changes on the secondary and will not be considered reliable in these rapidly changing conditions.
Tave is not accurate in these situations because it can change rapidly based on rod movement and difference between primary and secondary power.

ref. BGP 100-3 E. Limitation and Actions- 1. Reactor d. & h. (pgs 10 & 11)
states" Reactor Power to be monitored by all available indication to assure proper NI operation, ie Delta Temperature, turbine first stage pressure, generator output and steam pressure.

Question 67 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28310
User-Defined ID:	2012 NRC EXAM RO Q67
Cross Reference Number:	3E.AM-028-A
Topic:	2012 NRC Exam RO Question 67
Num Field 1:	R.O. 4.3
Num Field 2:	SRO 4.3
Text Field:	Gen. 2.1.45
Comments:	New Question, High Cog, RO level 2.1.45 **Ability to identify and interpret diverse indications to validate the response of |** **another indication. |** (CFR: 41.7 / 43.5 / 45.4) | IMPORTANCE RO 4.3 SRO 4.3

Associated objective(s):

Given an Industry Event, NRC, or INPO Communique', EVALUATE and APPLY the Lessons Learned, Good Operating Practices, or Safe Operating Attitudes to this station

Given:

- Unit 1 had been on line for 1 year of continuous full power operation.
- The reactor tripped 14 hours ago.
- A Unit 1 reactor startup has commenced.
- The crew has stabilized reactor power at 1×10^{-3}%, but the power ascension is delayed for one hour.
- In order to maintain reactor power constant at 1×10^{-3}%, you will have to operate controls that affect plant reactivity.

Either a ________ must be performed or the control rods must be ________ during the one-hour period.

A. boration, inserted

B. boration, withdrawn

C. dilution, inserted

D. dilution, withdrawn

Answer: A

Answer Explanation:

Meets K/A. Question involves a reactor start-up and requires examinee knowledge of control manipulation that affects reactivity.

Requires examinee to understand that Xenon will be "burning out" faster than it is being produced based on the stated time frame (peaked at about 10 hours and is now decaying). As the poison concentration is reduced it will have to be off-set by either inserting control rods or performing a boration.
All other distractors are plausible based on the examinee's knowledge of how and when xenon changes, following a reactor trip.

Ref: BCB-1 fig. 8c

Question 68 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28311
User-Defined ID:	2012 NRC EXAM RO Q68
Cross Reference Number:	3C.GP-03-I
Topic:	2012 NRC Exam RO question 68
Num Field 1:	RO 4.5
Num Field 2:	SRO 4.4
Text Field:	Gen2.2.1
Comments:	Bank Question, High Cog **2.2 Equipment Control** 2.2.1 **Ability to perform pre-startup procedures for the facility, including operating** **| those controls associated with plant equipment that could affect reactivity.** | (CFR: 41.5 / 41.10 / 43.5 / 43.6 / 45.1) IMPORTANCE RO 4.5 SRO 4.4

Associated objective(s):

ANALYZE a given power history which may include Startup, Power Increase(s), Power Decrease(s), or Shutdown and DETERMINE the changes in Xenon Concentration versus Time

Considering the Feedwater system differences between the Units:

- Which unit has a Loss of Tempering Line Subcooling Abnormal Operating procedure?
- Which unit has a flow and purge permissive logic circuitry to OPEN the Feed Water Isolation Valves (FW009A-D)?

Unit that has:

	Loss of Tempering Line Subcooling procedure	Flow and Purge Permissive
A.	1	1
B.	1	2
C.	2	1
D.	2	2

Answer: D

Answer Explanation:

Meets K/A. Requires examinee to know the procedurally based difference between manipulations at the feedwater panel of each unit.
Reference:
2BOA SEC-6, Loss of feedwater Tempering Line Subcooling has no comprable U-1 abnormal operating procedure
2BGP 100-3 step 40 has the actions necessary to satisfy the flow and purge permissives to open the FWIVs whereas on U-1 these restrictions do not exist.

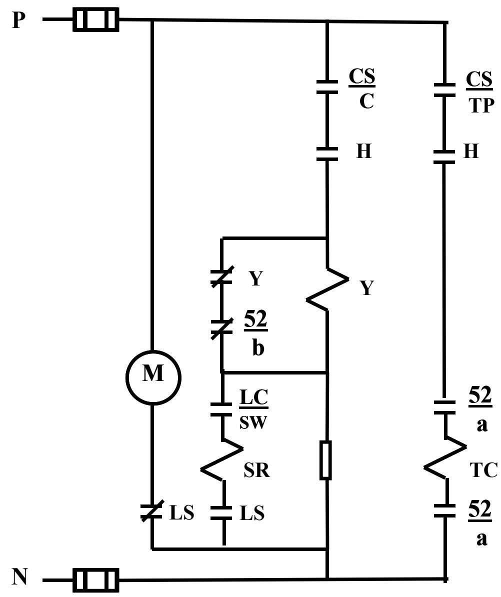
All distracters are plasible if the examinee does not know the differences between feedwater operation on each unit.

Question 69 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28312
User-Defined ID:	2012 NRC EXAM RO Q69
Cross Reference Number:	3C.FW-04-A
Topic:	2012 NRC Exam RO Question 69
Num Field 1:	RO 3.6
Num Field 2:	SRO 3.6
Text Field:	Gen. 2.2.4
Comments:	New Question, Low Cog 2.2.4 (multi-unit license) **Ability to explain the variations in control board/control room** | **layouts, systems, instrumentation, and procedural actions between units at a** | **facility.** | (CFR: 41.6 / 41.7 / 41.10 / 45.1 / 45.13) IMPORTANCE RO 3.6 SRO 3.6

Associated objective(s):

Given a set of plant conditions or parameters DETERMINE if the FW System is responding correctly.

Given the typical 4KV/6.9KV Breaker Electrical Schematic control circuit below:



Which contact listed below prevents the breaker from closing MORE than once after an initial closing signal is generated?

A. 52/b

B. LC/SW

C. Y

D. LS

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee ability to interpret station electrical drawings.

The circuit shown is a simplified example of the anti-pumping circuit used in all Byron 4KV and 6.9 KV breakers. All the contact options are in the circuit leg with the spring release (SR) coil, therefore any of the contacts can interrupt the circuit continuity to the SR coil.

Choice C is correct, the Y (anti-pumping) contact opens whenever the Y coil is energized (i.e. when the breaker has a active closing signal). The open Y contact will prevent the SR coil from getting a second closing signal (from a continuous close signal) in the event the breaker closed and tripped back open immediately. In that case the Y coil would remain energized through the resistor downstream of the Y coil, thereby keeping the Y contact open and preventing the breaker from reclosing after the closing spring had re-charged.
Choice B is incorrect, the 52/b (breaker position) contact is in the circuit as a protective device that will prevent a closing signal from energizing the SR coil on a breaker that is already closed (thus protecting the SR coil from prolonged energization and coil damage).

Choice D is incorrect, the LC/SW (latch check switch) contact is is a contact in a microswitch that senses the position of the breaker trip coil plunger, thereby preventing a closing signal from energizing the SR coil on a breaker that has an active trip signal in.
Choice A is incorrect, the LS (limit switch) contact is a limit switch that senses the charge state of the closing spring. Its function is to prevent a closing signal from energizing the SR coil on a breaker that does not have the closing spring charged and to energize the closing spring charging motor

Ref: 6E-1-4030B

Question 70 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28265
User-Defined ID:	2012 NRC EXAM RO Q70
Cross Reference Number:	A.BP3-03-A
Topic:	2012 NRC Exam RO Question 70
Num Field 1:	RO 3.5
Num Field 2:	SRO 3.9
Text Field:	Generic 2.2.41
Comments:	Bank Question from 2011 Braidwood NRC Exam High Cog. RO Level 2.2.41 **Ability to obtain and interpret station electrical and mechanical drawings.** | (CFR: 41.10 / 45.12 / 45.13) | IMPORTANCE RO 3.5 SRO 3.9

Associated objective(s):

Given an electrical schematic for a motor control circuit or breaker, EXPLAIN the following operation: Starting or Shutting

In accordance with EP-AA-113, Attachment 1, Emergency Worker Exposure Limits, a 30 year old Equipment Operator can receive a dose up to ________ Rem TEDE to protect valuable property?

A. 5

B. 10

C. 30

D. 60

Answer: B

Answer Explanation:

Meets K/A. Requires examinee knowledge of normal and emergency exposure limits.

5 Rem is the limit for "all" emergency activities. 30 Rem is the limit to the lens of the eye for protecting property. 60 Rem is plausible based on the formula Max. Dose = 5(N-18). For a 30 year old this would equal 60.

Ref: EP-AA-113 Att. 1

Question 71 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	3
Difficulty:	2.00
System ID:	28087
User-Defined ID:	CERT 2010 RO 97
Cross Reference Number:	T.AM46-03
Topic:	Emergency exposure limits
Num Field 1:	3.2
Num Field 2:	3.7
Text Field:	G2.3.04
Comments:	Bank Question, Low Cog, RO Level References EP-AA-113, Attachment 1 10CFR5543(b)(4) **2.3 Radiation Control** 2.3.4 **Knowledge of radiation exposure limits under normal or emergency conditions.** | (CFR: 41.12 / 43.4 / 45.10) | IMPORTANCE RO 3.2 SRO 3.7

Associated objective(s):

Byron ILT 09-1 Cert Exam

Given:

- Both units are at full power, normal alignment.
- The 0A VC train is in normal operation.
- The 0B VC train is in standby.

The following occurs:

- An event that has the potential for an accidental radioactive release in the Unit 2 Auxiliary Building is reported to the MCR.
- The MCR SRO directs an RO to monitor control room intake air for elevated radiation trends.
- The RO notes all MCR rad monitor icons on the RM-11 GRID 2, PROCESS AIR MONITORS, are currently GREEN.

With the above conditions, to monitor control room intake air on the RM-11, the RO will trend the...

A. 0PR31J or 0PR32J, OUT AIR IN OA

B. 0PR33J or 0PR34J, OUT AIR IN OB

C. 0PR35J or 0PR36J, TURB AIR IN OA

D. 0PR37J or 0PR38J, TURB AIR IN OB

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to use radiation monitoring systems.

MCR rad monitor icons are green even when their sampled plenums are not online because the sample pumps will continuously sample plenums that have stagnant air flow. With the 0A VC system in normal alignment (outside air intake) the only rad monitors that would have MCR intake air flow through their respective intake plenum is the 0PR31J and 32J.

Choice A is correct, see explanation above.
Choice B is incorrect, 0PR33J and 34J sample the outside air intake from Unit 2 (0B train). Although the radiation spill was in the Aux. bldg, because 0B VC train was not running, this plenum would not experience intake air flow.
Choice C is incorrect, 0PR35J and 36J sample the turbine bldg intake from Unit 1 (0A train). Although the radiation spill was in the Aux. bldg, this plenum would not experience intake air flow unless the 0A VC system was manually or automatically swapped to emergency mode.
Choice D is incorrect, 0PR37J and 38J sample the turbine bldg intake from Unit 2 (0B train). Although the radiation spill was in the Aux.bldg this plenum would not experience intake air flow unless the 0B VC system was manually started in emergency mode.

Ref: Horse Note AR-1a and BOP VC-1 C. Prerequisites (pgs. 2 and 3)

Question 72 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28263
User-Defined ID:	2012 NRC EXAM RO Q72
Cross Reference Number:	S.AR1-18
Topic:	2012 NRC Exam RO Question 72
Num Field 1:	RO 2.9
Num Field 2:	SRO 2.9
Text Field:	Generic 2.3.5
Comments:	Bank Question 2011 Brwd NRC exam Low Cog RO level 2.3.5 **Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.** | (CFR: 41.11 / 41.12 / 43.4 / 45.9) | IMPORTANCE RO 2.9 SRO 2.9

Associated objective(s):

Given a set of plant conditions, ANALYZE how these conditions are affected by any instrumentation, control circuit, or electrical power failure

After receiving a report of a fire in the plant, the Assist NSO will inform plant personnel that a fire exists by ALL of the following methods EXCEPT:

A. calling the Fire Marshall on the telephone.

B. announcing the fire over the radio.

C. actuating the Plant Wide Fire Alarm.

D. announcing the fire over the plant PA system.

Answer: A

Answer Explanation:

Meets the KA, requires knowledge of the examinee on how to respond to a plant fire.

All answers are plausible as they either are required to be done or there is good plausibility that it could be done.
All required actions are listed in BAP 1100-10

Ref: Bap 1100-10 page 3

Question 73 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28357
User-Defined ID:	2012 NRC EXAM RO Q73
Cross Reference Number:	3E.AM-011-A
Topic:	2012 NRC Exam RO Question 73
Num Field 1:	RO 3.3
Num Field 2:	SRO 3.7
Text Field:	Gen. 2.4.25
Comments:	Bank Question, Low Cog, RO level **2.4 Emergency Procedures / Plan** 2.4.25 **Knowledge of fire protection procedures.** | (CFR: 41.10 / 43.5 / 45.13) | IMPORTANCE RO 3.3 SRO 3.7

Associated objective(s):

DISCUSS the NSO's response upon the report of an In-Plant Fire

Given:

? A fire is occurring in the Upper Cable Spreading Room directly above the Unit 1 MCR.
? All detectors in the area are in alarm on 1PM09J, Fire Detection panel.
? The system responds as designed.

The operators first response, per 1PM09J Alarm Response Procedures, regarding fire suppression to this event is:

A. Dispatch an operator to locally actuate Halon suppression.

B. Verify automatic CO_2 suppression actuation.

C. Verify automatic Halon suppression actuation.

D. Dispatch an operator to locally actuate CO2 suppression.

Answer: C

Answer Explanation:

Meets K/A. Requires examinee to have knowledge of "fire in the plant" procedures, specifically the BARs in this case.

The correct answer is correct base on the Alarm Response procedure.

Manual Halon actuation is plausible but only if auto action does not occur.
The room also has back-up CO2 suppression but will not actuate automatically, making the CO2 distractors plausible

Ref: BAR 1PM09J-E1 and E2

Question 74 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28358
User-Defined ID:	2012 NRC EXAM RO Q74
Cross Reference Number:	S.FP1-06-C
Topic:	2012 NRC Exam RO question # 74
Num Field 1:	RO 3.4
Num Field 2:	SRO 3.9
Text Field:	Gen. 2.4.27
Comments:	Bank Question, Low Cog, RO level 2.4.27 **Knowledge of "fire in the plant" procedures.** | (CFR: 41.10 / 43.5 / 45.13) | IMPORTANCE RO 3.4 SRO 3.9

Associated objective(s):

STATE the plant areas protected by the following: Halon

During an Emergency Event when control has been transferred from the Control Room to the fully staffed Emergency Response Organization, how is an assigned Safe Shutdown EO dispatched to perform an "in plant task"?

A. The EO reports to the MCR and is dispatched by the Unit Supervisor.

B. The WEC Supervisor contacts the EO via the radio from the WEC.

C. The MCR contacts the TSC who contacts the OSC. The OSC dispatches the EO.

D. The EO reports to the TSC and is dispatched by the Emergency Operations Director.

Answer: A

Answer Explanation:

Meets KA. Requires examinee to have knowledge of techniques to use to obtain EO support during an Emergency event when the ERO is staffed.

Per EP-AA-112-200-F-18. The Shift Manager will track teams dispatched by the Control Room prior to OSC activation, until they are released to the OSC. The Safe shutdown EO's not going to be processed through the OSC per the last bullet of the pre-defined Teams. The safe shut down EO's are dispatched by the control room.

All distractors are plausible based on examinee knowledge of the process.

Ref: EP-AA-112-200-F-18

Question 75 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28347
User-Defined ID:	2012 NRC EXAM RO Q75
Cross Reference Number:	3F.ZP-04-A
Topic:	2012 NRC Exam RO Question 75
Num Field 1:	RO 3.2
Num Field 2:	SRO 3.8
Text Field:	Gen. 2.4.43
Comments:	Bank Question, Low Cog. 2.4.43 **Knowledge of emergency communications systems and techniques**. | (CFR: 41.10 / 45.13) IMPORTANCE RO 3.2 SRO 3.8

Associated objective(s):

DISCUSS the proper technique of communications with off-site agencies when acting as communicator and the responsibilities of a "Communicator"

Given:

A stuck open pressurizer safety valve on Unit 1 has resulted in the following:

? Reactor Trip and Safety Injection
? All ECCS systems are running as designed
? Pzr Level is off scale high
? RVLIS indicates 0% in the head
? RCPs are OFF
? RCS Subcooling is NOT acceptable
? CETCs indicate 600-605 degrees F and stable
? PRT pressure is 50 psig and RISING
? PRT temperature is 135 degrees F and RISING
? The crew has completed the initial diagnosis steps in 1BEP-0 Reactor Trip or Safety Injection, but is still in the procedure.

Which of the following procedures will the SRO implement to address this event?

A. 1BEP ES-0.0, REDIAGNOSIS.

B. 1BFR-I.3, RESPONSE TO VOIDS IN THE REACTOR VESSEL.

C. 1BFR-I.1, RESPONSE TO HIGH PRESSURIZER LEVEL.

D. 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

Answer: D

Answer Explanation:

Meets KA. Requires examinee to assess the effects of a bubble in the reactor vessel due to a stuck open safety valve and determine the correct recovery procedure. This is an SRO question because it has the examinee assess plant conditions, then select the appropriate recovery procedure.

Although the circumstances described in the question place the examinee in 1BEP-0 at or past the step where the commencement of the monitoring of the Status Trees occurs, neither of the two BFR procedures offered as distracters will address the problem. They are plausible because the question indicates a void exists in the vessel (RVLIS is 0 in the head), and pressurizer level is off scale high.
The first step of the I.1 and the I.3 procedures check if ECCS flow has been terminated, and if it has not, then return to procedure and step in effect. The question indicates that ECCS has not been terminated because the ‘ECCS pumps are operating as designed’, and ‘Subcooling is NOT acceptable’ which prohibits reducing ECCS injection flow.
The Rediagnosis distractor is plausible because the off normal PRT indications have no direct transistion to 1BEP-1, and the examinee that does NOT know the requirements to enter the Rediagnosis procedure may try to apply it here.
The correct answer is 1BEP-1 because the RISING PRT conditions will eventually rupture the PRT and cause the Rad and floor drain sump levels in containment to rise. This provides the transistion to 1BEP-1 from step 23 of BEP-0.

Ref: 1BEP-0 step 23 (pg 19)

Question 76 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28348
User-Defined ID:	2012 NRC EXM SRO Q76
Cross Reference Number:	T.FR06-03
Topic:	2012 NRC Exam SRO Question 76
Num Field 1:	
Num Field 2:	
Text Field:	
Comments:	New Question, High Cog., SRO level **APE: 008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)** **AA2. Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: (CFR: 43.5 / 45.13)** AA2.29 The effects of bubble in reactor vessel 3.9 4.2 43.b.5

Associated objective(s):

STATE the basis for the actions described in steps, notes, and cautions

Given:

- Unit 1 experienced a Large Break LOCA.
- A transition to 1BEP-1, "LOSS OF REACTOR OR SECONDARY COOLANT," has been made.
- Subsequently, 1BEP ES-1.3, "TRANSFER TO COLD LEG RECIRCULATION," was implemented.
- Currently, the operators are aligning ECCS for Cold Leg Recirculation per Step 2.

The STA reports a RED path in the Heat Sink status tree.

When will the actions of 1BFR H.1, LOSS OF SECONDARY HEAT SINK, be performed?

The actions of 1BFR H.1 will be commenced IMMEDIATELY after...

A. Step 3, Align RH Pumps Suction to Cnmt Sumps.

B. Step 4, Check SI and Cent Chg Pumps in ECCS Injection Mode.

C. Step 5, Align SI and Cent Chg Pumps for Cold Leg Recirculation.

D. Step 6, Start ECCS Pumps As Necessary.

Answer: D

Answer Explanation:

Meets KA. Requires examinee to coordinate Functional Restoration Procedures while implementing Emergency Procedures using implementation hierarchy. This is an SRO level question because it requires knowledge of when to implement attachments and other procedures in conjuction with the procedure that is in affect at the time.

The correct answer is correct based on a Note in 1BEP ES-1.3 that states to Perform Steps 1 thru 6 without delay.
All distractors are plausible based on examinees knowledge

Ref: 1BEP ES-1.3 pages 1-10 (note before step 1 and after step 6)

Question 77 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28349
User-Defined ID:	2012 NRC EXM SRO Q77
Cross Reference Number:	T.EP02-01-C
Topic:	2012 NRC EXAM SRO QUESTION 77
Num Field 1:	RO 3.5
Num Field 2:	SRO 4.4
Text Field:	011gen. 2.4.16
Comments:	New Q, Hi Cog. SRO level **011 Large Break LOCA** 2.4.16 **Knowledge of EOP implementation hierarchy and coordination with other** **| support procedures or guidelines such as, operating procedures, abnormal** **| operating procedures, and severe accident management guidelines.** | (CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 3.5 SRO 4.4

Associated objective(s):

Without the use of EP-1, EP ES-1.1, 1.2, 1.3, 1.4: EXPLAIN the purpose of each "note" and "caution" in the procedure

Given:

- Both Units are at 100% power.
- CC is in a normal alignment with Unit 0 CC HX and pump aligned to Unit1.

The U-0 CC HX is isolated due to a large leak.

What action, if any, is required?

A. NO Tech Spec action is required.

B. Enter LCOAR for Unit 1 ONLY.

C. Enter LCOAR for Unit 2 ONLY.

D. Enter LCOAR for BOTH Units.

Answer: D

Answer Explanation:

Meets K/A. Requires Examinee to determine operability status of CC system on each unit due to the U-0 CC HX leak. This is a SRO level question as it requires knowledge of the T/S bases section 3.7.7

Per the Tech. Spec Bases, "Since the CC system is shared, the common heat exchanger and associated portions of its flowpath, may be credited to both units. Further on "The inoperability of the common CC heat exchanger impacts both units' flow paths".

All distractors are plausible based on the examinees knowledge of the 7-day LCOAR. Ref. Bases, 3.7.7. Action A-1. (pg 5)

Question 78 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28318
User-Defined ID:	2012 NRC EX. SRO Q78
Cross Reference Number:	S.CC1-16
Topic:	2012 NRC Exam SRO Question 78
Num Field 1:	RO 2.9
Num Field 2:	SRO 3.5
Text Field:	026AA2.01
Comments:	Bank Question, Low Cog, SRO level **026 Loss of Component Cooling Water (CCW)** **AA2. Ability to determine and interpret the following as they apply to** **the Loss of Component Cooling Water:** **(CFR: 43.5 / 45.13)** AA2.01 Location of a leak in the CCWS 2.9 3.5 43.6.2

Associated objective(s):

Given applicable reference material, ANALYZE a given set of plant conditions and DETERMINE Component Cooling Water Tech Spec/TRM operability requirements.

Given:

The crew took conservative action to TRIP the reactor and manually initiate a Safety Injection due to RISING Containment Pressure. All Safety Systems responded as designed. 1BEP-0, REACTOR TRIP OR SAFETY INJECTION is being implemented by your crew.

Immediately following the reactor trip the following conditions are noted:

- PZR Level Lowering RAPIDLY
- PZR Pressure Lowering RAPIDLY
- Tave is 535 degress and continuing to Lower
- All S/G Levels Lowering RAPIDLY
- All SG Pressures Lowering RAPIDLY
- ALL MSIVs are currently OPEN
- Containment pressure is 4.7 psig and RISING

You are now in the diagnostic steps of BEP-0.

From the following selections, which procedure transition will be made NEXT?

A. 1BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

B. 1BEP-2, FAULTED STEAM GENERATOR ISOLATION.

C. 1BEP-3, STEAM GENERATOR TUBE RUPTURE.

D. 1BCA 2.1, UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS.

Answer: B

Answer Explanation:

Meets K/A. Requires examinee to distinguish between a Stm line break and a LOCA. SRO question due to detailed knowledge of the parameters that are required to make the transition to the appropriate recovery procedure.

SG pressure lowering in an uncontrolled manner is one of 2 kick outs to the E-2 procedure.
All distractors are plausible as they are all transitions from E-0 with the exception of BCA 2.1. BCA 2.1 is plausible based on depressurization of all s/g which is presented in the stem.

Ref: 1BEP-0 step 21 (pg 17)

Question 79 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28319
User-Defined ID:	2012 NRC EX SRO Q79
Cross Reference Number:	T.EP01-06-A
Topic:	2012 NRC Exam SRO question 79
Num Field 1:	RO 4.6
Num Field 2:	SRO 4.7
Text Field:	040AA2.03
Comments:	New Question, High Cog., SRO Level **040 Steam Line Rupture** **AA2. Ability to determine and interpret the following as they apply to** **the Steam Line Rupture:** **(CFR: 43.5 / 45.13/ 43.b.5)** AA2.03 Difference between steam line rupture and LOCA 4.6 4.7

Associated objective(s):

GIVEN a set of plant conditions, DIAGNOSE and ANALYZE a Reactor Trip or Safety Injection

Given:

- Unit 1 was in Mode 3 when a Loss of Offsite Power (LOOP) occurred.
- The Crew is performing 1BEP ES-0.4, Natural Circulation Cooldown with Steam Void in Vessel (Without RVLIS).

The following indications are available:

? Offsite Power has NOT been Restored
? ALL WR RCS Hot leg temperatures are 200-205 degrees F
? RCS pressure is 300 psig
? Shutdown cooling was established 30 hours ago
? All SG NR levels are 60%
? There is NO steam flow from any SG

What action will the SRO direct NEXT?

A. Start all CRDM exhaust and booster fans.

B. Continue to cooldown the RCS to below 200 degrees F.

C. Transition to 1BGP 100-5, Plant Shutdown and Cooldown.

D. Raise SG levels to 88% to condense any remaining steam in the SGs.

Answer: B

Answer Explanation:

The question meets the KA by testing the ability to make an operational judgment (what to do NEXT) for a loss of offsite power situation, given various plant indications.
The question is high cog, because of the need to compare the parameters given, with what is required, and make a choice based on the results. (Comprehension).
The question is SRO level because of the detailed knowledge of the procedure required to answer the question. This falls under 10CFR55.43.b.5 and the guidance provided by the NRC.
The conditions given in the stem put the examinee at step 23 of the ES-0.4 procedure. One of the requirements to depressurize the RCS at that point is that the entire RCS is below 200 degrees F. If this is not the case, then further cooldown of the RCS is necessary via the RH system in shutdown cooling (per step 21) which was performed in step 20.
The transition to 1BGP 100-5 distractor is plausible but not correct because that is the next action if all the conditions were met.
Starting all CRDM fans is plausible, but incorrect, because the procedure requires them to be running for the entire cooldown or the RH system in shutdown cooling for at least 27 hrs. With a loss of offsite power, the CRDM fans are not available, but RH has been on line for 30 hrs.
Raising SG levels to condense any steam left is plausible, but incorrect, because one of the requirements to depressurize the RCS at this point is that NO SG is still steaming.

Ref: 1BEP ES-0.4 step 20 through 23 (pg.15-17)

Question 80 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28350
User-Defined ID:	2012 NRC EXM SRO Q80
Cross Reference Number:	3D.OA-21-A
Topic:	2012 NRC Exam SRO Question 80
Num Field 1:	RO 4.4
Num Field 2:	SRO 4.7
Text Field:	056gen.2.1.7
Comments:	New question, High Cog, SRO Level **APE: 056 Loss of Offsite Power** **2.1.7 Ability to evaluate plant performance and make operational judgments based on |** **operating characteristics, reactor behavior, and instrument interpretation. |** (CFR: 41.5 / 43.5 / 43.b.5/ 45.12 / 45.13) | IMPORTANCE RO 4.4 SRO 4.7

Associated objective(s):

Given a set of plant conditions or parameters indicating a Loss of Offsite Power, PREDICT the integrated plant response to the event/casualty with no operator action

Given the following plant conditions on Unit 1:

? A reactor shutdown occurred 15 days ago for a refueling outage.
? RCS temperature is currently 122 °F.
? The core reload was completed and Mode 5 was entered 2 days ago.

Subsequently, the following events occurred:
? The 1A RH pump tripped on overcurrent.
? The 1B RH pump would NOT start.
? The crew performed an RCS bleed and feed using both PZR PORVs and the 1B CV pump.
? RCS pressure is 180 psig.
? RWST level is 91%.

Which of the following will provide the LEAST amount of injection flowrate, in gallons per minute, while preventing boiling?.

(REFERENCES PROVIDED)

A. 50

B. 90

C. 370

D. 535

Answer: C

Answer Explanation:

Meets K/A. Requires examinee to interpret curves contained in 1BOA Pri-10 to determine RCS make-up rate to prevent boiling. This is an SRO question because the examinee must assess plant conditions and then select the correct portion of the procedure to mitigate the event.

Requires examinee to determine which of the 2 reference graphs to use which makes all distractors plausible based on using the incorrect curve. Examinee must also use the correct curve to "prevent boiling" versus the curve to "match" boil off. Again making all distractors plausible.

Ref: 1BOA Pri-10 pages 11 and 12.

Question 81 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28321
User-Defined ID:	2012 NRC EX SRO Q81
Cross Reference Number:	3D.OA-09-D
Topic:	2012 NRC Exam SRO Question 81
Num Field 1:	RO 3.9
Num Field 2:	SRO 4.2
Text Field:	025G2.1.25
Comments:	Bank Question, High Cog, SRO Level **025 Loss of Residual Heat Removal System (RHRS)** 2.1.25 **Ability to interpret reference materials, such as graphs, curves, tables, etc.** | (CFR: 41.10 / 43.5/ 43.b.5 / 45.12) | IMPORTANCE RO 3.9 SRO 4.2

Associated objective(s):

Given a set of plant conditions or parameters indicating a Loss of RH Cooling and a set of plant procedures, IDENTIFY the correct procedure(s) to be utilized and DISCUSS required operator actions

Given:

- Unit 1 is at full power.
- A Routine containment vent release is in progress.

A review of the RM-11 has identified the following information:

? 1AR011 indicates 2.80 E+01 mr/hr
? 1AR012 indicates 0.00 E+00 mr/hr

A ten minute average review of 1AR12 on the RM-11 identified that approx 20 minutes ago, 1AR012 was indicating a steady 3.10 E+01 mr/hr.
The associated RM-23 currently indicates the same as the RM-11.

Which, if any, of the following is the TECH SPEC. REQUIRED action?
(REFERENCES PROVIDED)

A. Immediately secure the unit 1 CNMT vent release.

B. Restore 1AR012 within 4 hours, otherwise isolate the affected penetration within the next 1 hour.

C. Restore 1AR012 within 4 hours, otherwise isolate the affected penetration within the next 4 hours.

D. No Action Required.

Answer: C

Answer Explanation:

Meets K/A, requires examinee to analyze a failure of an AR instrument and determine the required Tech Spec. actions for that failure. Question encompasses failure of instrumentation (section 3) and containment isolation (section 6) of Tech. Specs.This is SRO level because the examinee needs to apply the required actions associated with the Tech Specs.

Application of Condition A of LCO 3.3.6, 4 hours to restore rad monitor, then to Condition B which requires implementing the requirements of LCO 3.6.3. Use of Bases table and associated RM-11 BAR requires application of Condition A allowing 4 hours to isolate penetration.

Question also requires knowledge of the VQ Release path valves.

All distractors are plausible based on examinees knowledge of the VQ system and Tech. Specs.

Ref: Tech. Spec 3.3.6 and 3.6.3 LCO and bases

Question 82 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28322
User-Defined ID:	2012 NRC EX SRO Q82
Cross Reference Number:	3C.AR-02-B
Topic:	2012 NRC Exam SRO Question 82
Num Field 1:	RO 3.2
Num Field 2:	SRO 4.1
Text Field:	061AA2.06
Comments:	Bank Question, High Cog. SRO Level **061 Area Radiation Monitoring (ARM) System Alarms** **AA2. Ability to determine and interpret the following as they apply to** **the Area Radiation Monitoring (ARM) System Alarms: (CFR: 43.5 / 45.13/43.b.2)** AA2.06 Required actions if alarm channel is out of service 3.2 4.1

Associated objective(s):

Given a set of plant conditions or parameters involving the radiation monitoring system, DETERMINE Tech Spec compliance and required actions

Consider Section 3.1, Reactivity Control, of the Technical Specifications.

Tech. Spec. 3.1.1- Shutdown Margin AND
Tech. Spec. 3.1.8- Physics Tests exceptions-Mode 2
Require INITIATION of boration within 15 minutes if SHUTDOWN MARGIN is NOT met.

YET

Tech. Spec. 3.1.4- Rod Group Alignment Limits
Tech. Spec. 3.1.5- Shutdown Bank Insertion Limits AND
Tech. Spec. 3.1.6- Control Bank Insertion Limits
Require INITIATION of boration within 1 hour if their limits are NOT met.

What is the reason for the difference in these time frames when they all are concerned with SHUT DOWN MARGIN?

It allows time for the Operator to....

A. to REALIGN rods.

B. to determine if rods are TRIPPABLE.

C. perform verification of SHUTDOWN MARGIN.

D. to REDUCE power to below applicable limits.

Answer: C

Answer Explanation:

Meets K/A, requires knowledge of the SRO to determine boration requirements associated with SDM that are contained in the bases documents. This is an SRO question concerning as it requires the examinee to analyze T/S bases.

The reason is correct as listed in the bases documents.
All distractors are plausible as they are also contained in the bases.

Ref: B 3.1.1 (pg 4) B 3.1.4 (pg 7)

Question 83 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28351
User-Defined ID:	2012 NRC EXM SRO Q83
Cross Reference Number:	S.CV2-09
Topic:	2012 NRC Exam SRO Question 83
Num Field 1:	RO 3.2
Num Field 2:	SRO 4.2
Text Field:	024 gen. 2.2.25
Comments:	New Question, Low Cog, SRO level **024 Emergency Boration** 2.2.25 **Knowledge of the bases in Technical Specifications for limiting conditions for** | **operations and safety limits.** | (CFR: 41.5 / 41.7 / 43.2/ 43.b2) IMPORTANCE RO 3.2 SRO 4.2

Associated objective(s):

DESCRIBE the applicable Technical Specifications bases concerning the RMCS

Given:

An event occurs that results in a unit trip from full power. A feedwater isolation signal malfunction occurred at the time of the trip that has resulted in ALL Steam Generator Levels being off-scale HIGH. The crew is currently in BEP ES-0.1 REACTOR TRIP RESPONSE and the Shift Manager has directed the STA to monitor the Status Trees.

Current major plant indications are as follows:

- IR SUR: -0.3 DPM
- RCS Tave 530 degrees F and stable
- PZR Level 15% and stable
- RCS Pressure 1900 psig and slowly rising
- All SG pressures >1235 psig
- All SG Levels off-scale HIGH

The Shift Manager directs you to commence a plant cooldown to Mode 5 conditions as soon as possible.

From the selections below, choose the Yellow Path Procedure that will be entered FIRST, to accomplish the Shift Managers direction.

A. S.2, LOSS OF CORE SHUTDOWN.

B. H.2-RESPONSE TO SG OVERPRESSURE.

C. I.2, RESPONSE TO LOW PRESSURIZER LEVEL.

D. H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL.

Answer: B

Answer Explanation:

Meets K/A requires prioritizing yellow paths due to a steam generator overpressurization event. This is an SRO question because it requires the examinee to assess plant conditions and then select a procedure to mitigate the event/comply with the Shift Managers request.

The overpressurization event is caused by overfilling the SGs. You can not directly enter into BFR H-3 response to SG high level. You must first enter BFR H-2, response to SG overpressure, which will direct you to BFR H-3. The reason the H series is the highest priority currently is because you can not draw steam down the steam lines with greater than 91% level in the SGs. In order to cool down, draining SG level will be your highest priority.

Per BAP 1310-10, Procedure Use and Adherence, Byron Addendum, Yellow Path implementation is based on operator judgement when it is determined that adequate time exsists to implement it. "In other words, the operator does not have to implement the procedure". The stem of the question states the Shift Manager desires to cool the plant down to Mode 5. To do that the S/G level will have to be lowered, which will result in lowering SG pressure and allow steam removal without adversely affecting the steam piping. The S-2 BST is a higher priority but does not have to be implemented because SUR criteria is not met. The correct answer is to perform the plant cooldown IAW SM direction and to do that BFR H.2 becomes the Unit Supervisors priority.

Ref: 1BST-3 and BAP 1310-10 pg 20

BFR S-2 is considered a plausible distractor because during a cooldown positive reactivity will be added to the core.
BFR I-2 is considered a plausible distractor because it may be anticipated PZR level will drop further as the cooldown commences.
Yellow path BFR's are taught that there is no immediate or prompt concern with their implementation which makes this distractor plausible.

Question 84 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28324
User-Defined ID:	2012 NRC EX SRO Q84
Cross Reference Number:	T.FR03-04-B
Topic:	2012 NRC Exam SRO Question 84
Num Field 1:	RO 3.6
Num Field 2:	SRO 4.4
Text Field:	E13.2.4.22
Comments:	New Question, High Cog. SRO Level **E13 Steam Generator Overpressure** 2.4.22 **Knowledge of the bases for prioritizing safety functions during |** **abnormal/emergency operations. |** (CFR: 41.7 / 41.10 / 43.5 / 43.b.5/ 45.12) | IMPORTANCE RO 3.6 SRO 4.4

Associated objective(s):

Given a set of plant conditions, DIAGNOSE and ANALYZE a Response to Steam Generator Overpressure

Given the following sequence of events:

? U1 Reactor Trip and SI occurred from a normal 100% power lineup.
? 1BEP-0 "Reactor Trip or Safety Injection" is entered.
? ALL U1 ECCS equipment is verified to be operating properly.
? While performing 1BEP-0 step 24, "Check if ECCS Flow Should be Terminated", the crew notes the following U1 indications:

S/G Pressures	ALL 1080 psig and stable
Containment Pressure	0.8 psig and stable
Pzr Pressure	2000 psig and stable
Containment Rad Levels	Normal
Containment Floor Water Level	0"
S/G NR levels	ALL 35% and slowly RISING
RCS Temperature	554°F and slowly RISING
Pzr Level	19% and slowly RISING
PRT Pressure	54 psig and RISING
Pzr PORVs	CLOSED

With the above conditions, the NEXT procedure the crew will transition to is...

A. 1BEP ES-1.1 "SI TERMINATION" and then to 1BEP ES-1.2 "POST LOCA COOLDOWN and DEPRESSURIZATION"

B. 1BEP ES-1.1 "SI TERMINATION" and then to 1BEP-1 "LOSS OF REACTOR OR SECONDARY COOLANT".

C. 1BEP-1 "LOSS OF REACTOR OR SECONDARY COOLANT" and then to 1BEP ES-1.2 "POST LOCA COOLDOWN and DEPRESSURIZATION".

D. 1BEP-1 "LOSS OF REACTOR OR SECONDARY COOLANT" and then to 1BGP 100-5 "PLANT SHUTDOWN and COOLDOWN".

Answer: A

Answer Explanation:

Meets K/A. Requires examinee to analyze current conditions as applied to a LOCA and then select the correct procedure to transition to based on the conditions. This is an SRO question based on the previous statement.

If we did not transition from 1BEP-0 during the diagnostic steps (steps 21-23), step 24 of E-0 checks if ECCS flow should be reduced. The criteria are: Subcooling acceptable (given in the stem), secondary heat sink available as indicated total feedflow and SG level (given in the stem), RCS pressure stable or rising (given in the stem) and adequate RCS inventory, PZR level, (given in the stem). This meets the criteria for transistioning to ES1.1 SI Termination. Once charging pumps are realigned, RCS pressure will again drop, sending the crew to 1BEP ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION. The SI pumps will not be injecting based on RCS pressure. When ECCS flow is reduced to 1 charging pump going through the normal charging header and Pzr level or RCS pressure begins to fall per procedure, transition to ES 1.2.

All other distractors are plausible based on the examinees mis-diagnosis of the event or other misconception.

Ref: 1BEP-0 step 24 (pg. 20)
and 1BEP ES-1.1 step 5 (pg. 3)

Question 85 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28325
User-Defined ID:	2012 NRC EX SRO Q85
Cross Reference Number:	T.EP02-09-C
Topic:	2012 NRC Exam SRO Question 85
Num Field 1:	RO 3.4
Num Field 2:	SRO 4.2
Text Field:	E03EA2.1
Comments:	Bank Question, High Cog, SRO Level **E03 LOCA Cooldown and Depressurization** **EA2. Ability to determine and interpret the following as they apply to** **the (LOCA Cooldown and Depressurization)** (CFR: 43.5 / 43.b.5/ 45.13) EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations. IMPORTANCE RO 3.4 SRO 4.2

Associated objective(s):

GIVEN a set of plant conditions, DIAGNOSE and ANALYZE a Post LOCA Cooldown and Depressurization

Given:

- The unit is at full power in a normal full power alignment.
- The crew has entered 1BOA PRI-12, UNCONTROLLED DILUTION based on RISING RCS temperatures.

- The RO reports that 1TI-130, LTDWN HX OUTLET TEMP, is reading 78 degrees.

Which of the following actions is contained in 1BOA PRI-12 and will mitigate this situation?

A. isolate letdown.

B. take manual control of 1TK130, LTDWN HX OUT TEMP CONTROL 1CC130, and RAISE controller output.

C. place 1CV129, DEMIN HI TEMP LTDWN DIVERT VLV, to the VCT position.

D. place 1CV112A, LTDWN TO VCT OR HUT DIVERT VALVE, to the HUT.

Answer: C

Answer Explanation:

Meets KA. Requires examinee to analyze an event and based on that, be familiar with what parameters are beyond limits. This requires detailed knowledge of BOA PRI-12 RNO actions. As such this is an SRO question because of the knowledge required of dianostic steps and decision points contained within the procedure.

The correct answer is correct based on procedure direction to direct the flow stream around the dilution component (i.e demins).
1CV112A divert to the HUT is plausible but not contained in the procedure.
1CC130A action is plausible but in this case is going in the incorrect direction.
Isolating Letdown is plausible but not contained within the procedure.

Ref. 1BOA Pri-12 step 3 RNO (pg 4)

Question 86 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28359
User-Defined ID:	2012 NRC EXM SRO Q86
Cross Reference Number:	T.OA22-02
Topic:	2012 NRC EXAM SRO QUESTION 86
Num Field 1:	RO 4.2
Num Field 2:	SRO 4.3
Text Field:	004A2.06
Comments:	New Question, High Cog. SRO level **004 Chemical and Volume Control System** **A2 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:** (CFR: 41.5/ 43/5 / 43.b.5/ 45/3 / 45/5) A2.06 Inadvertent boration/dilution 4.2 4.3

Associated objective(s):

ANALYZE a given set of plant conditions and DETERMINE if entry into 1/2BOA PRI-12, Uncontrolled Dilution, is required

Unit 2 is in the process of a plant cooldown with the following conditions:

- 2B RH train is in the shutdown cooling mode.
- RCS temperature is 300°F.
- RCS pressure is 340 psig.
- 2B CV pump is in operation
- PZR level is 40% and Saturated.

From the list below, identify the (1) malfunction and (2) the mitigating procedure that will address the malfunction that will cause the GREATEST LOWERING of the 2B RH Hx Outlet temperature.

	(1) Malfunction	(2) Procedure to correct malfunction
A.	2CC9412B, CC TO RH HX 2B ISOL VLV, fails OPEN	2BOA SEC-4, LOSS OF INSTRU. AIR
B.	2RH607, HX 2B FLOW CONT VLV, fails OPEN AIR	2BOA SEC-4, LOSS OF INSTRU.
C.	2RH607, HX 2B FLOW CONT VLV, fails OPEN	2BOA ELEC-5, LOCAL EMERGENCY CONTROL OF SAFE SHUTDOWN EQUIP.
D.	2CC9412B, CC TO RH HX 2B ISOL VLV, fails OPEN	2BOA ELEC-5, LOCAL EMERGENCY CONTROL OF SAFE SHUTDOWN EQUIP.

Answer: B

Answer Explanation:

Meets K/A. Requires examinee to understand "support" system failure and how they affect the RH System. Also requires examinee to choose a procedure that will help mitigate the failure making this an SRO question.

The correct answer is correct because on a loss of air the RH HX outlet valve will fail open allowing maximum RH flow through the heat exchanger. The bypass valve will throttle down to maintain total flow of 3300 gpm. The correct procedure would restore instrument air

2CC9412B failing open tests the examinees knowledge of the system. It is a MOV and will normally be full open while in Shutdown Cooling. If the examinee feels this valve is a throttled AOV that fails open would make this distractor plausible. The Local Emergency Control of safe shutdown equipment is a plausible distractor but does not address the air failure. It does address operation of an MOV.

Ref: BOA Sec-4 Table A (pg 13)

Question 87 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28327
User-Defined ID:	2012 NRC EX SRO Q87
Cross Reference Number:	S.RH1-11
Topic:	2012 NRC Exam SRO Question 87
Num Field 1:	RO 2.7
Num Field 2:	SRO 2.9
Text Field:	005A2.01
Comments:	Bank Question, High Cog. SRO Level **005 Residual Heat Removal System (RHRS)** **A2 Ability to (a) predict the impacts of the following malfunctions or** **operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of** **those malfunctions or operations:** (CFR: 41.5 / 43.5 /43.b.5/ 45.3 / 45.13) A2.01 Failure modes for pressure, flow, pump motor amps, motor temperature, and tank level instrumentation 2.7 2.9*

Associated objective(s):

Given a set of plant conditions, ANALYZE those conditions and PREDICT how they are affected by any RH System instrumentation, control circuit, or electrical power failure

Given:

- ? The Unit experienced a Large Break LOCA 5 minutes ago from full power.
- ? All systems responded as designed.

The Reactor Operator reports to you that the following annunciators are LIT on panel 6.

- ? Annunciator _-6-A2, RH PUMP AUTO START
- ? Annunciator _-6-A1, RH PUMP TRIP
- ? 1B RH pump control switch has an amber disagreement light

Which ONE of the below actions will you direct the crew to take based on the current annunciator status?

	Place the 1A control switch in:	Place the 1B control switch in:
A.	after close	leave "as is"
B.	after close	pull out
C.	leave "as is"	pull out
D.	leave "as is"	leave "as is"

Answer: B

Answer Explanation:

Meets KA. Requires examinee knowledge of Alarm Response Procedures of the ECCS components and the SRO to provide direction of control board operation of those components, making this an SRO question.

The correct answer is correct based on BAR 1-6-A1 (trip) and 1-6-A2 (auto start) actions. Leave "as is" is considered plausible, however is not in accordance with the procedures.

Ref: BAR 1-6-A1 and A2

Question 88 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28360
User-Defined ID:	2012 NRC EXM SRO Q88
Cross Reference Number:	T.EP02-01-D
Topic:	2012 NRC Exam SRO Question 88
Num Field 1:	RO 4.2
Num Field 2:	SRO 4.0
Text Field:	006gen2.4.50
Comments:	New Question, Low Cog, SRO level **006 Emergency Core Cooling System (ECCS)** **2.4 Emergency Procedures / Plan (continued)** || 2.4.50 **Ability to verify system alarm setpoints and operate controls identified in the** | **alarm response manual.** | (CFR: 41.10 / 43.5 / 43.b.5/ 45.3) | IMPORTANCE RO 4.2 SRO 4.0

Associated objective(s):

Without the use of EP-1, EP ES-1.1, 1.2, 1.3, 1.4: EXPLAIN the intent of and basis for each step in the procedure

Given:

- 2BOA PRI-1, EXCESSIVE PRIMARY PLANT LEAKAGE, was entered due to a partially OPEN pressurizer PORV.
- Manual isolation was attempted but was UNSUCCESSFUL.
- A Reactor Trip and Safety Injection actuation has occurred.

With the EOPs entered, when will the actions of 2BOA PRI-1 be continued/completed?

2BOA PRI-1 actions...

A. will be resumed IMMEDIATELY after the completion of the Immediate Actions Steps of 2BEP-0, REACTOR TRIP OR SAFETY INJECTION.

B. IMMEDIATELY upon transition OUT OF 2BEP-0.

C. will NOT be completed.

D. will be resumed after 2BEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, is exited.

Answer: C

Answer Explanation:

Meets K/A. Requires examinee to have knowledge of how BOAs are used in conjuction with BEPs. SRO level based on directing the crews procedural transitions.

The correct answer is correct because the BOAs will not be re-entered, unless specifically called out in a EP, CA or FR procedure after entry into a BEP has been performed. This is deliniated in BAP 1310-10. Further as a general "rule of thumb" the OA's are applicable only until the reactor is tripped or SI actuated.

All distractors are plausible based on the examinees knowledge of these procedural directions.

Ref: BAP 1310-10 page 10

Question 89 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28352
User-Defined ID:	2012 NRC EXM SRO Q89
Cross Reference Number:	T.OAQ12-02
Topic:	2012 NRC Exam SRO Question 89
Num Field 1:	RO 3.8
Num Field 2:	SRO 4.5
Text Field:	007G2.4.8
Comments:	New question, High Cog, SRO level **007 Pressurizer Relief Tank/Quench Tank System (PRTS)** 2.4.8 **Knowledge of how abnormal operating procedures are used in conjunction with |** **EOPs. |** (CFR: 41.10 / 43.5 /43.b.5/ 45.13) | IMPORTANCE RO 3.8 SRO 4.5

Associated objective(s):

ANALYZE a given set of plant conditions and DETERMINE if entry into 1/2BOA PRI-1, Excessive Primary Plant Leakage, is required

Given:

- Unit 1 reactor was at full power.
- 1A SX pump is OOS.
- A reactor trip occurred and 1BEP-0 "REACTOR TRIP OR SI" was entered.
- A Loss of All AC Power occurred two minutes later and BOTH D/Gs did NOT automatically start.
- 1A D/G was manually started at step 5 of 1BCA-0.0, LOSS OF ALL AC POWER.
- 1A D/G output breaker automatically closed and re-energized bus 141.
- 1B D/G could NOT be started.
- An EO reports 1A DG local alarm 1PL07J-1-C2, ESSENTIAL SERVICE WATER FLOW LOW is LIT solid.

Under these conditions, the next ACTION the SRO takes is to…

A. transition to 1BEP-0, REACTOR TRIP OR SAFETY INJECTION.

B. transition to 1BOA ELEC-3, LOSS OF 4KV ESF BUS.

C. remain in 1BCA-0.0, dispatch an operator to emergency stop the DGs and perform steps to cross tie bus 142 to bus 242

D. transition to 1BOA PRI-7, ESSENTIAL SERVICE WATER MALFUNCTION.

Answer: C

Answer Explanation:

Question meets KA - question requires examinee predict the impacts of loss of SWS malfunction on the SX system and determine correct actions to mitigate the consequences of malfunction. Prior to transitioning out of 1BCA-0.0 in step 5, the procedure checks to ensure DG support systems are energized. In this case the 1A DG did not have SX pp support, so transitioning out of CA-0.0 is incorrect transition. CA-0.0 continues to cross tie ESF buses with opposite unit according to which train has DG support equipment available. In this case bus 142 has SX pump available so it will be crosstied to unit 2 in CA-0.0. This is an SRO level question because the examinee is required to assess the conditions and then select the procedure or portion of procedure to mitigate the event.

A is incorrect, this would leave the 1A DG running without cooling for an extended period of time.
C is correct, the 1A DG is running without cooling so it is immediately stopped. Further action in 1BCA 0.0 will cross tie the ESF busses with U-2 and get the 1B SX pump running to support DG cooling on U-1.
D is incorrect, this would leave the 1A DG running without cooling for an extended period of time.
B is incorrect, this is improper transition from CA-0.0.

Ref: BCA 0.0 step 5 (pg. 5 and 6)

Question 90 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28260
User-Defined ID:	12 NRC EXAM SRO Q90
Cross Reference Number:	7D.CA-001A
Topic:	2012 NRC Exam SRO Question 90
Num Field 1:	RO 3.5
Num Field 2:	SRO 3.7
Text Field:	076A2.01
Comments:	From Brwd 2011 Cert Exam Hi Cog SRO Level **076 Service Water System (SWS)** **A2 Ability to (a) predict the impacts of the following malfunctions** **or operations on the SWS; and (b) based on** **those predictions, use procedures to correct, control,** **or mitigate the consequences of those malfunctions or operations:** (CFR: 41.5 / 43.5 / 43.b.5/ 45/3 / 45/13) A2.01 Loss of SWS 3.5* 3.7

Associated objective(s):

DISCUSS the basic steps of the CA-0 series procedures

Given:

- Unit 1 is in Mode 3.
- 1D RCP is running.
- Operators are preparing to close ALL Reactor Trip Breakers and energize ALL Control Rod Drive Mechanisms.

Based on the above conditions, a second RCS loop must be placed in operation to…

A. reduce core neutron leakage by acting as a reflector.

B. ensure homogeneous boron concentration throughout the core.

C. ensure adequate core decay heat removal.

D. mitigate the postulated power excursion from accidental control rod withdrawal.

Answer: D

Answer Explanation:

Question meets KA - question requires examinee knowledge of the bases in Tech Specs.. Is SRO level based on the bases and application of the required actions.
TS 3.4.5 bases requires two RCS loops in operation to ensure postulated accidents associated with a power excursion from inadvertent control rod withdrawal is mitigated.
A is incorrect, reason is for RCS loops in operation in Modes 1 and 2. Reactor is subcritical in Mode 3.
B is incorrect, reason is for one RCS loop in operation. One RCS loop will provide adequate boron mixing.
D is correct, see explanation above.
C is incorrect, reason is for one RCS loop in operation. One RCS loop will provide adequate decay heat removal.

Ref T/S B 3.4.5 pages 1 and 2.

Question 91 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28281
User-Defined ID:	2012 NRC EXAM SRO 91
Cross Reference Number:	S.RC1-14
Topic:	2012 NRC Exam SRO Question #91
Num Field 1:	RO 3.2
Num Field 2:	SRO 4.2
Text Field:	002 2.2.25
Comments:	Bank Question from 2009 Brwd Cert. exam Low Cog **002 Reactor Coolant System** 2.2.25 **Knowledge of the bases in Technical Specifications for limiting conditions for** | **operations and safety limits.** | (CFR: 41.5 / 41.7 / 43.2/ 43.b.2) IMPORTANCE RO 3.2 SRO 4.2

Associated objective(s):

DISCUSS the bases for the Reactor Coolant System Tech Specs

A startup is in progress following a reactor trip that occurred 24 hours ago. The following plant conditions exist:

? Unit is in MODE 2.
? Shutdown Banks are all withdrawn.
? Control Bank D is 50 steps withdrawn.
? Rx power is stable at 1.0 E-3% power.

THEN:

? A fault on SAT 142-1 occurs, which results in a loss of DRPI and Rod Bottom Lights.

The NEXT 2 procedures entered, IN ORDER, are:

A. BFR-S.1 RESPONSE TO NUCLEAR POWER GENERATION/ATWS, to BEP-0 REACTOR TRIP OR SAFETY INJECTION.

B. BFR-S.1 RESPONSE TO NUCLEAR POWER GENERATION/ATWS, to BEP ES-0.2 NATURAL CIRCULATION COOLDOWN.

C. BEP-0, REACTOR TRIP OR SAFETY INJECTION, to BGP 100-5 PLANT SHUTDOWN AND COOLDOWN.

D. BEP-0 REACTOR TRIP OR SAFETY INJECTION, to ES-0.1 REACTOR TRIP RESPONSE.

Answer: D

Answer Explanation:

Meets KA. Questions examinee on Loss of offsite power, states that DRPI will be lost in this situation, unit not on line, and requires examinee to use the correct procedures to mitigate the consequences of the loss of offsite power. SRO question requiring examinee to assess plant conditions and then select the appropriate 2 procedures which will next be entered.

The correct answer is correct because a fault on either SAT will result in both SATs deenergizing resulting in a loss of all AC Buses until the DGs start and supply the ESF buses. Per page 1 of E-0 under symptoms of a reactor trip, a rapid drop is neutron level indicated by nuclear instrumentation will be exhibited and this procedure should be invoked. Since a Safety Injection condition is not indicated in the question stem the proper transition is to ES-0.1 Reactor trip response.
BFR S.1 is considered plausible based on the misconception that the examinee has lost indication of reactor power due to ROD Bottom lights and DRPI not being available.
Go to BGP 100-5 is considered plausible but to get there you first transition to BEP ES-01. If NO RCPs are running, as is the case with the question stem, then entry to BEP ES-0.2 is the correct path but again you need to go to ES-0.1 first.

Ref: 1BEP-0 Symptoms or Entry Conditions, 1BEP ES-0.1 Nat. Circ. Cooldown and step 14 (kick out to ES-0.2)

Question 92 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28330
User-Defined ID:	2012 NRC EX SRO Q92
Cross Reference Number:	T.OA04-12
Topic:	2012 NRC Exam SRO Question 92
Num Field 1:	R.O 2.8
Num Field 2:	SRO 3.3
Text Field:	014A2.01
Comments:	Bank Question, High Cog, SRO Level **014 Rod Position Indication System (RPIS)** **A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RPIS; and (b) based on those on those predictions, use procedures to correct, control,** **or mitigate the consequences of those malfunctions or operations**: (CFR: 41.5 / 43.5 /43.b.5/ 45.3 / 45.13) A2.01 Loss of offsite power 2.8 3.3

Associated objective(s):

DESCRIBE the actions necessary to stabilize the plant following the Loss of Offsite Power

Given

The 1A DG is running for the monthly surveillence.
A heat detector malfunction in the 1A DG room has resulted in a CO_2 actuation.

The CO_2 actuation _____(1)______ cause the diesel to TRIP.
This event is ____(2)_____.

A. (1) will NOT
(2) Reportable

B. (1) will
(2) NOT Reportable

C. (1) will
(2) Reportable

D. (1) will NOT
(2) NOT Reportable

Answer: A

Answer Explanation:

Meets K/A. Requires examinee to analyze a faulty fire protection detector and what effect that will have on a running DG. The question further asks about the reportability manual to make this an SRO question.

The correct answer is correct based on the fact that the DG will continue to run and that the event is classified as reportable in accordance with LS-AA-1110. The detector failure causes CO_2 actuation, creating a hazardous environment which is entry to either a UE or an AI. Any EAL is reportable.

All distractors are plausible based on examinee knowledge.

Ref: LS-AA-1110 Reportable event SAF 1.12 (pg. 46) or 1.13 (pg 48) or SAF 1.23 (pg. 70)

Question 93 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28362
User-Defined ID:	2012 NRC EXM SRO Q93
Cross Reference Number:	8E.AM-102
Topic:	2012 NRC Exam SRO question 93
Num Field 1:	RO 2.7
Num Field 2:	SRO 2.9
Text Field:	086A2.03
Comments:	New Question, High Cog, SRO level **086 Fire Protection System (FPS)** **A2 Ability to (a) predict the impacts of the following malfunctions** **or operations on the Fire Protection System;** **and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:** (CFR: 41.5 / 43.5 / 45.3 / 45.13) A2.03 Inadvertent actuation of the FPS due to circuit failure or welding 2.7 2.9

Associated objective(s):

SCREEN Reportable or Significant Events for Reportability

Per BAP 1310-10, HU-AA-104-101, PROCEDURE USE AND ADHERENCE, BYRON ADDENDUM, which of the following correctly describes when an emergency procedure action on the Operator Action Summary (OAS) page, is applicable?

A. ANY time during the applicable procedure performance, unless a specific procedural starting point is referenced in the action.

B. Only PRIOR to performing the applicable step in the main body of the procedure.

C. Only after proceeding PAST the applicable step in the main body of the procedure, BUT it will NEVER apply after a transition is made to another procedure.

D. Only after proceeding PAST the applicable step in the main body of the procedure, AND it MAY apply after a transition is made to another procedure.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to interpret and execute procedure steps. Examinee must know rules of usage for operator action summary actions in order to properly execute them.
The question is SRO level because SROs read and direct the emergency procedure actions and have the responsibility of monitoring the OAS actions as the procedure is being performed. Although ROs are responsible for the content of the OAS, the SRO is responsible for knowing the rules of usage and when to implement OAS actions.
BAP 1310-10 states that a step on the OAS page contains information that must be monitored throughout the procedure.

Choice B is incorrect, OAS actions apply as soon as the procedure is entered.
Choice D is incorrect, however, this may be applicable to continuous action summary steps depending on the action and whether it is superseded or no longer applicable to the next procedure.
Choice C is incorrect, however, this may be applicable to continuous action summary steps depending on the action and whether it is superseded or no longer applicable to the next procedure.
Choice A is correct, see explanation above

Ref: BAP 1310-01 pg. 11 and 15

Question 94 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28264
User-Defined ID:	12 NRC EXAM SRO Q94
Cross Reference Number:	T.AM04-16
Topic:	2012 NRC Exam SRO Question 94
Num Field 1:	RO 4.6
Num Field 2:	SRO 4.6
Text Field:	Generic 2.1.20
Comments:	Bank Question (2011 Brwd NRC Exam) Low Cog, SRO Level **2.1 Conduct of Operations** 2.1.20 **Ability to interpret and execute procedure steps.** | (CFR: 41.10 / 43.5 /43.b.5 / 45.12) | IMPORTANCE RO 4.6 SRO 4.6

Associated objective(s):

DESCRIBE how to use the Operating Department procedures

Given:

A Special Test is to be conducted for the Electrical Engineering Group.

From the below list, what are the Operation's Department procedural responsibilities associated with conducting this test?

1. Writes the test procedure
2. Approves the implementation of the test
3. Leads the HLA/IPA
4. Attends the HLA/IPA
5. Attends the PJB
6. Ensures the plant conditions are maintained during the test
7. Coordinates the test

A. 2, 5 and 7

B. 2, 4 and 6

C. 4, 6 and 7

D. 1, 3 and 5

Answer: B

Answer Explanation:

Meets K/A. Tests examinee on the process for conducting special or infrequent tests. This is SRO level knowledge due to approval of the test which is an operations department function.

Per OP-AA-108-110 Operations Shift Management (section 3.4) approves the implementation of the test, ensures the plant conditions are maintained during the test and (section 4.3.3) attends the HLA/IPA.

All other distractors are plausible based on examinee's knowledge of the procedure.
Coordination is done by the test coordinator
Test procedure is developed by the testing group, in this case engineering
PJB's are not performed for these activities
The HLA/IPA will be conducted by senior line management or designee

Ref: OP-AA-108-110 pgs. 2 and 4

Question 95 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28332
User-Defined ID:	2012 NRC EX SRO Q95
Cross Reference Number:	T.AM23-01-B
Topic:	2012 NRC Exam SRO Question 95
Num Field 1:	RO 2.9
Num Field 2:	SRO 3.6
Text Field:	Gen. 2.2.7
Comments:	New Question, Low Cog., SRO level **2.2 Equipment Control** 2.2.7 **Knowledge of the process for conducting special or infrequent tests.** | (CFR: 41.10 / 43.3 / 45.13) IMPORTANCE RO 2.9 SRO 3.6

Associated objective(s):

DESCRIBE the activities that require a HLA Briefing

Given:

- Unit 1 is at full power, in a normal full power alignment.
- You are the Unit 1 Unit Supervisor.
- Annunciator 1-21-D6, 125V DC BUS 111 GROUND, ALARMED 10 minutes ago.
- You have dispatched an Equipment Operator to investigate in accordance with the Alarm Response Procedure.
- The Equipment Operator reports back that the Bus 111 Ground Detector is reading +118 volts to ground.

Choose the correct answer concerning the (1) priority of the work and (2) whether to enter 1BOL DC-1, LCOAR, ESF BUS DC GROUNDS?

	(1) Priority	(2) Enter 1BOL DC-1
A.	B1	yes
B.	B3	no
C.	B1	no
D.	B3	yes

Answer: A

Answer Explanation:

Meets KA, Requires knowledge of procedures for work prioritization. SRO level due to determining LCOAR entry.

Per 1BOL DC1 grounds greater than 115 volts are B-1 priority. The BOL is entered at greater than or equal to 75 volts to ground.
All distractors are plausible based on examinee's knowledge.

Ref: BAR 1-21-D6 and 1BOL DC1 pg.3

Question 96 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28333
User-Defined ID:	2012 NRC EX SRO Q96
Cross Reference Number:	7E.AM-138-A
Topic:	2012 NRC Exam SRO Question 96
Num Field 1:	RO 2.6
Num Field 2:	SRO 3.8
Text Field:	Generic 2.2.17
Comments:	New Question, Low Cog, SRO Level 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.** | (CFR: 41.10 / 43.5 / 45.13) | IMPORTANCE RO 2.6 SRO 3.8

Associated objective(s):

DESCRIBE the proper way to prioritize the required maintenance on shift

Given:

- The 0E Waste Gas Decay Tank has been sampled by Chemistry and reported to contain 3.8 E+5 Curies.
- The tank pressure is 75 psig.

Your required action AND mitigating strategy to control the release associated with this Waste Gas Decay Tank will be...

	Action	Mitigating Strategy
A.	Enter 0BOA RAD-3, DECAY TANK HIGH ACTIVITY	pressurize WGDT to 95 psig with nitrogen gas
B.	Isolate WGDT	let WGDT decay for 30 days
C.	Enter 0BOA RAD-3, DECAY TANK HIGH ACTIVITY	transfer some of the tank contents to another WGDT
D.	place standby WGDT on line	perform a release of 0E WGDT

Answer: C

Answer Explanation:

Meets KA by requiring examinee knowledge of entry condition of BOA (RO Level) and assessing plant conditions and implementing a mitigating strategy (SRO level) for high activity associated with a waste gas decay tank.

The correct answer is correct per the procedure.
Pressurizing the WGDT with N2 is considered plausible based on N2 connections to the Waste Gas System and pressurizing the system may be a misconception to dilute the tank contents.
Isolating the tank is a plausible distractor if the examinee is not aware of entry conditions to the BOA, letting the tank decay for 30 days is also plausible.
Placing another WGDT in standby and performing a release of the 0E WGDT is plausible if the candidate believes there is no problem.

Ref: 0BOA RAD-3 step 2 (pg 2)

Question 97 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28334
User-Defined ID:	2012 NRC EX SRO Q97
Cross Reference Number:	T.OA26-03
Topic:	2012 NRC Exam SRO Question 97
Num Field 1:	RO 3.8
Num Field 2:	SRO 4.3
Text Field:	Gen. 2.3.11
Comments:	New Question, Low Cog, SRO Level **2.3 Radiation Control** 2.3.11 **Ability to control radiation releases.** | (CFR: 41.11 / 43.4 / 43.b.5/ 45.10) | IMPORTANCE RO 3.8 SRO 4.3

Associated objective(s):

ANALYZE a given set of plant conditions and DETERMINE the required actions per 0BOA RAD-3, Decay Tank High Activity

- Unit 1 is mode 2, with a Unit start up in progress.
- 1BGP 100-2A1, REACTOR STARTUP, has just been completed.
- An emergent activity requires a containment entry.

In accordance with BAP 1450-1, ACCESS TO CONTAINMENT, which one of the following Unit 1 activities must the US ensure does NOT occur during the containment entry?

A. A mode change to Mode 3.

B. A mode change to Mode 1.

C. A containment release.

D. An RCFC fan swap.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of radiological safety procedures pertaining to containment entry.
The question is SRO level because mode changes are authorized by the SM and require SRO concurrence.
BAP 1450-1, Access To Containment, step 4.3.1.1 restricts operations from changing modes to a higher power level with personnel in containment. However, changing modes to a lower power level is acceptable.
Choice B is correct, see explanation above.
Choice A is incorrect, see explanation above.
Choice D is incorrect, a containment release does not have any restrictions during containment entry.
Choice C is incorrect, a RCFC swap does not have any restrictions during containment entry.

Ref: BAP 1450-1 step 4.3.1 (pg 4)

Question 98 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28266
User-Defined ID:	12 NRC EXAM SRO Q98
Cross Reference Number:	T.AM17-01
Topic:	2012 NRC Exam SRO Question 98
Num Field 1:	R.O. 3.4
Num Field 2:	SRO 3.8
Text Field:	Generic 2.3.13
Comments:	Bank Question from Braidwood 2011 NRC Exam Low Cog. SRO Level 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.** | (CFR: 41.12 / 43.4 / 45.9 / 45.10) IMPORTANCE RO 3.4 SRO 3.8

Associated objective(s):

DISCUSS the requirement for containment entries during shutdown and power operations

Given:

- The crew is currently staffed with the minimum number of required qualified fire brigade members per BAP 320-1, SHIFT STAFFING.
- Six hours into a twelve hour shift, a fire brigade member must leave work unexpectedly.

Under these conditions, to meet the requirements of BAP 320-1, the SM/designee...

A. does NOT need to call out a replacement BECAUSE the Fire Brigade is allowed one unexpected absence for a PARTIAL shift.

B. must take IMMEDIATE action to call out a replacement AND have the position filled within 2 hours.

C. does NOT need to call out a replacement BECAUSE the Fire Brigade Chief can fill the member's role during an unexpected absence for a PARTIAL shift.

D. must take action WITHIN 2 HOURS to call out a replacement AND have the position filled as soon as possible after that.

Answer: B

Answer Explanation:

The question meets the KA, requires knowledge of the fire brigade requirements. The question is SRO level because maintaining the minimum shift staffing is the responsibility of the SM (SRO) and is normally delegated to the WEC supervisor during emergent backshift situations.

Per BAP 320-1, step C.2, Fire Brigade composition may be less than the minimum requirements for a period not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the vacancy.

All distractors are plausible based on the examinee's knowledge of this procedure.

Ref: BAP 320-1 (pg. 4)

Question 99 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28354
User-Defined ID:	2012 NRC EXM SRO Q99
Cross Reference Number:	7E.AM-057-A
Topic:	2012 NRC Exam SRO Question 99
Num Field 1:	RO 3.1
Num Field 2:	SRO 3.6
Text Field:	Gen. 2.4.26
Comments:	Bank Question, Low Cog, SRO level **2.4 Emergency Procedures / Plan** 2.4.26 **Knowledge of facility protection requirements, including fire brigade and |** **portable fire fighting equipment usage. |** (CFR: 41.10 / 43.5 / 45.12) | IMPORTANCE RO 3.1 SRO 3.6

Associated objective(s):

Given a set of Shift Manpower Sheets, EVALUATE required shift manning

The Tech Spec limits for the amount of stored diesel fuel oil that is required to be maintained on site is based upon having sufficient supply for each diesel generator to supply...

Note:
LOCA - Loss of Coolant Accident
LOOP - Loss of Off Site Power

A. 3 days of post LOCA load demand.

B. 7 days of post LOCA load demand.

C. 14 days of post LOOP shutdown load demand.

D. 30 days of post LOOP shutdown load demand.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to explain system limits.
Tech Spec 3.8.3 bases states the DG are supplied with enough stored oil for 7 days of post LOCA loads.
The question is SRO level because it requires knowledge of Tech Spec bases.

Choice D is incorrect, 3 days (72 hours) is action completion time for TS 3.8.1 qualified circuit.
Choice B is correct, see explanation above.
Choice C is incorrect, 14 days is action completion time for TS 3.8.1 DG.
Choice A is incorrect, 30 days is action completion time for TS 3.8.3 fuel oil properties out of tolerance.

Ref: B 3.8.3 LCO portion (pg. 2)

Question 100 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	0
Difficulty:	0.00
System ID:	28355
User-Defined ID:	2012 NRC EX SRO Q100
Cross Reference Number:	S.DG1-12
Topic:	2012 NRC Exam SRO Quest. 100
Num Field 1:	RO 3.8
Num Field 2:	SRO 4.0
Text Field:	gen. 2.1.32
Comments:	Bank question (2011 Bwd NRC Exam SRO portion) Low cog, sro level **2.1 Conduct of Operations** 2.1.32 **Ability to explain and apply system limits and precautions.** | (CFR: 41.10 / 43.2 / 45.12) IMPORTANCE RO 3.8 SRO 4.0

Associated objective(s):

DISCUSS the bases for the Diesel Generator Tech Specs