LICENSEE PROPOSED 2009 EXAM RO-Q1-75, SRO-Q76-100

1 **ID: RS20003-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power.
- The VCT hydrogen pressure regulator is stuck closed.
- All other systems are normally aligned.
- Over several shifts VCT pressure drops to 10 psig.

The effect of lowering VCT pressure on the operation of RCP seals is...

A. rising flow through the #2 RCP seal.

B. rising flow through the #3 RCP seal.

C. lowering flow through the #2 RCP seal.

D. lowering flow through the #1 RCP seal.

Answer: C

Answer Explanation:

Question meets K/A- requires examinee to know cause effect relationship between RCP system and CVCS (VCT).

With < 15 psig (reduced backpressure on #1 Seal), #1 Seal leakoff flow will rise, thus causing less flow through the #2 Seal.
Choice A is incorrect, opposite of answer.
Choice B is incorrect, the effect of lowering #2 seal flow would cascade up to the #3 seal and lower flow through #3 seal also.
Choice C is correct, see explanation above.
Choice D is incorrect, opposite would be true. Lower pressure on LP side of #1 seal would raise seal flow through #1 seal.

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:	11056
User-Defined ID:	RS20003-N01
Cross Reference Number:	BWLI-RC2-030
Topic:	RS20003-N01
Num Field 1:	2.6
Num Field 2:	2.9
Text Field:	SYS003 K1.04
Comments:	Braidwood NRC Exam 2009 Question # 1 003 Reactor Coolant Pump Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems: CVCS TIER: 2 GROUP: 1 Cog Level: High Task No: R-RC-006 Obj No: 4C.RC-05 Cross Ref: 10CFR55.41(b)(3)

General Data	
Technical Reference with Revision Number:	BwOP RC-1 rev. 22 BwOP CV-13 rev. 22 ILT LP I1-RC-XL-02 rev. 4b
Justification for Non SRO CFR Link:	
Question History:	Not currently selected on any exam
Question Source:	Bwd ILT Bank
Low KA Justification (if required):	
Revision History:	Revised stem and one distractor to ensure it is incorrect.
ILT	
Supplied Ref (If appropriate): (i.e. ABN-##)	None
Excluded Reference: (i.e. Ensure ON-## not provided)	

2 **ID: RS20004-N01** **Points: 1.00**

Given:
- Unit 1 is in MODE 2, reactor start up in progress.
- All U-1 systems are normally aligned for this mode.
- Unit 2 is at 100% power, normal alignment.

The following sequence of events occurs:
- Unit 1 offsite power is LOST.
- All equipment operates as designed.
- The crew is performing step 1 of 1BwEP ES-0.1, REACTOR TRIP RESPONSE after determining SI was NOT required .
- At the "CHECK RCS TEMPERATURES" step it is determined that emergency boration is required per 1BwOA PRI-2, EMERGENCY BORATION.
- Non-ESF buses are still de-energized.
- NO other operator actions have been taken.

With the above conditions, the NSO will Borate the RCS from the ...

A. BAST via 1CV8104, EMER BORATION VALVE, and normal charging header.

B. BAST via 1CV110A, BORIC ACID TO BLNDR VLV and 1CV110B, BORIC ACID BLNDR TO CHG PMPS VLV and normal charging header.

C. RWST via 1CV112D and/or 1CV112E, RWST TO CHG PUMP SUCT VLVS and normal charging header.

D. RWST via 1CV112D and/or 1CV112E, RWST TO CHG PUMP SUCT VLVS and high head injection header.

Answer: C

Answer Explanation:

Question meets K/A - requires examinee knowledge of power supplies to RMCS make up pump (boric acid pump)
BA pump is powered from a non-ESF bus, so it is not available for flow through the normal boration flowpath or through CV8104. Non-ESF buses lost power when LOOP occurred in mode 2. Do not get restored until later in 1BwEP ES-0.1 Gravity feed through CV112D & E (ESF powered) from the RWST is the only option. Normal charging header should be available since all equipment operates as designed and 1BwOA PRI-2 directs boration via the normal charging header prior to attempting it through the high head injection header.
Choice A is incorrect, no power to boric acid pump
Choice B is incorrect, no power to boric acid pump
Choice C is correct, see explanation above
Choice D is incorrect, see explanation above

Question 2 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11057
User-Defined ID:	RS20004-N01
Cross Reference Number:	BWLI-CV2-020
Topic:	RS20004-N01
Num Field 1:	2.9
Num Field 2:	3.1
Text Field:	SYS004 K2.02
Comments:	Braidwood NRC Exam 2009 Question # 2 004 Chemical and Volume Control System Knowledge of bus power supplies to the following: Makeup pumps TIER: 2 GROUP: 1 Cog Level: High Task No: R-OA-033 Obj No: 3D.OA-08-E Cross Ref: 10CFR55.41(b)(6)

General Data	
Technical Reference with Revision Number:	1BwOA PRI-2 rev. 101
Justification for Non SRO CFR Link:	
Question History:	Used in ILT systems
Question Source:	ILT Bank
Low KA Justification (if required):	
Revision History:	revised stem and distractor for 09 NRC exam.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

3 **ID: RS20005-N01** **Points: 1.00**

Given the following sequence of events:
- Unit 1 was at 100% power, normal alignment.
- An RCS LOCA and auto SI occur.
- 1A RH pump tripped and will NOT re-start.
- All other equipment functioned as designed.

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

With the 1A RH pp unavailable, the cold leg recirc alignment status of the U1 CV pumps will be...

A. ONLY the 1B CV pump running with supply from SI Pumps suction cross-tie header via 1SI8924 SI AND CV PMPS SUCT HDR XTIE VLV.

B. BOTH CV pumps running with supply from 1B RH Pump discharge via 1CV8804A RH HX 1A TO CV PMP SUCT ISOL VLV.

C. BOTH CV pumps running with supply via 1CV112D/E RWST TO CHG PMPS SUCT VLV.

D. BOTH CV pumps running with supply from 1B RH Pump discharge via 1SI8804B RH HX 1B TO SI PMP SUCT ISOL VLV.

Answer: D

Answer Explanation:

Question meets K/A - requires examinee knowledge of how 1A RH pump malfunction will effect ECCS system (CV pps).
Per 1BwEP 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 would be opened by the procedure. Both CV pumps remain running.
Choice A is incorrect, both CV pumps will be running. Would be plausible answer if a single RH pump did not have capacity to supply both SI and CV pumps.
Choice B is incorrect, 1CV8804A will be closed and 1RH8716A/B will be closed (flowpath will be isolated)
Choice C is incorrect, RWST suction path will be isolated.
Choice D is correct, see explanation above

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:	11058
User-Defined ID:	RS20005-N01
Cross Reference Number:	RS10006-003
Topic:	RS20005-N01
Num Field 1:	3.7
Num Field 2:	3.8
Text Field:	SYS005 K3.05
Comments:	Braidwood NRC Exam 2009 Question # 3 005 Residual Heat Removal System (RHRS) Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: ECCS TIER: 2 GROUP: 1 Cog Level: High Task No: R-EP-015 Obj No: 3D.EP-14-B Cross Ref: 10CFR55.41(b)(7)

General Data	
Technical Reference with Revision Number:	LP I1-EC-XL-01 rev. 4c 1BwEP ES-1.3 rev. 200 ECCS-1 big note rev. 9
Justification for Non SRO CFR Link:	
Question History:	Previous version used in ILT COPS Phase
Question Source:	Modified from Bwd ILT Bank #RS10006-003
Low KA Justification (if required):	
Revision History:	Modified stem from bus 142 fault to 1A RH pump fault. Modified answer and distractors to require status of pumps and suction flow paths
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

4 **ID: RS20006-N01** **Points: 1.00**

Regarding the ECCS design:
10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems, lists the maximum calculated fuel element cladding temperature shall NOT exceed...

A. 700°F.

B. 2200°F.

C. 4800°F.

D. 5080°F.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of effect that a loss of ECCS could have on the fuel. Requiring the examinee to know the design criteria of ECCS allows the K/A to be examined in a discriminating fashion.

Choice A is incorrect, 700°F at CETCs is the degraded core cooling criteria.
Choice B is correct, 10 CFR 50.46 lists 2200°F as the design acceptance criteria for maximum clad temp.
Choice C is incorrect, 4800°F is the melting temperature of fuel at EOL.
Choice D is incorrect, 5080°F is the melting temperature of unirradiated UO_2.

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:	11076
User-Defined ID:	RS20006-N01
Cross Reference Number:	
Topic:	RS20006-N01
Num Field 1:	4.3
Num Field 2:	4.4
Text Field:	SYS006 K3.02
Comments:	Braidwood NRC Exam 2009 Question # 4 006 Emergency Core Cooling System (ECCS) Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: Fuel TIER: 2 GROUP: 1 Cog Level: Low Task No: R-SI-005 Obj No: 3S.SI-01 Cross Ref: 10CFR55.41(b)(7)

General Data	
Technical Reference with Revision Number:	10 CFR 50.46 ILT Lesson Plan I1-EC-XL-01 rev. 4c
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

5 **ID: RS20007-N01** **Points: 1.00**

With the Unit 1 reactor at full power, the Unit NSO notices a rising PRT level. In addition to the PZR PORVs and PZR Safety valves, the component(s) that could be causing PRT level rise is/are the ...

A. Letdown Line Relief Valve.

B. #2 RCP Seal Leakoff lines.

C. SI Accumulator Drain Valve.

D. #3 RCP Seal Leakoff lines.

Answer: A

Answer Explanation:

Question meets K/A - Requires examinee knowledge of which components discharge to the PRT.
Choice A is correct, letdown line relief discharges to the PRT.
Choice B incorrect, #2 seal leak off is directed to the RCDT outlet line and is isolated from the PRT by 1RY8031 (normally closed valve).
Choice C incorrect, SI Accumulator drain header is directed to the RCDT.
Choice D incorrect, #3 seal leak off is directed to the containment floor drain sump.

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:	11059
User-Defined ID:	RS20007-N01
Cross Reference Number:	BWLC3CRY2001
Topic:	RS20007-N01
Num Field 1:	2.7
Num Field 2:	2.9
Text Field:	SYS007 A3.01
Comments:	Braidwood NRC Exam 2009 Question # 5 007 Pressurizer Relief Tank/Quench Tank System (PRTS) Ability to monitor automatic operation of the PRTS, including: Components which discharge to the PRT TIER: 2 GROUP: 1 Cog Level: Low Task No: R-RY-013 Obj No: 3C.RY-02-B Cross Ref: 10CFR55.41(B) (3)

General Data		
Technical Reference with Revision Number:	P & ID M-60 sht. 6	Revision #:
Justification for Non SRO CFR Link:		
Question History:	Not used on any exams	
Question Source:	LORT Bank	
Low KA Justification (if required):		
Revision History:		
ILT		
Supplied Ref (If appropriate):	None	
Excluded Reference:		

6 **ID: RS20007-N02** **Points: 1.00**

Given:
- Unit 1 is in Mode 2 at 3% reactor power.
- All systems are normally aligned for this power level.

The following occurs:
- A pressurizer safety valve sticks open.
- Pressurizer pressure is dropping.
- Pressurizer level is rising.

With the above conditions and NO operator action, the red First Out annunciator reactor trip signal will be...

A. 1-11-C3 PZR PRESS LOW RX TRIP

B. 1-11-B4 OT DT RX TRIP

C. 1-11-A3 PZR LEVEL HIGH RX TRIP

D. 1-11-C1 PZR PRESS LOW SI/RX TRIP

Answer: D

Answer Explanation:

Question meets K/A - Requires examinee to predict impact of stuck open safety valve on PRZ system. Question only test part (a) of K/A IAW NUREG ES-401 section D.2.a (page 6) because part (b) does not create a discriminating question.
At 5% power (< P-7), the PRZ low pressure and high level trips are blocked. The OT DT trip is active, however at 5% power the DT is so low and the OT DT setpoint is so high, that it does not come close to reaching the setpoint even with lowering pressure penalty applied.
Choice A is incorrect, trip is blocked.
Choice B is incorrect, see explanation above.
Choice C is incorrect, trip is blocked.
Choice D is correct, see explanation above.

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:	11060
User-Defined ID:	RS20007-N02
Cross Reference Number:	BWLI-RP2-080
Topic:	RS20007-N02
Num Field 1:	3.9
Num Field 2:	4.2
Text Field:	SYS007 A2.01
Comments:	Braidwood NRC Exam 2009 Question # 6 007 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: Stuck-open PORV or code safety TIER: 2 GROUP: 1 Cog Level: High Task No: R-EP-032 Obj No: 3D.EP-02-A Cross Ref: 10CFR55.41(B) (5)

General Data		
Technical Reference with Revision Number:	I1-RP-XL-02 rev. 2a, ILT Reactor Protection lesson plan.	Revision #:
Justification for Non SRO CFR Link:		
Question History:	Not currently used in ILT	
Question Source:	Modified from Bwd ILT bank #BWLI-RP2-080	
Low KA Justification (if required):		
Revision History:	Changed stem of question from RY spray valve failing open to RY safety valve failing open.	
ILT		
Supplied Ref (If appropriate):	None	
Excluded Reference:		

7 **ID: RS20008-N01** **Points: 1.00**

Given:
- Unit 1 is in Mode 4.
- RCS cooldown commenced 2 hrs. 30 minutes ago.
- 1B RH Train is in Shutdown Cooling Mode.
- All systems are normally aligned for this mode.

The following annunciator alarms in the MCR:
- 1-2-C5, CC HX OUTLET TEMP HIGH

The proper response to the alarm is throttle OPEN...

A. 1CC9503, SPENT FUEL PIT HX 1 OUTLT CC FCV.

B. 1CC9507B, RH HX 1B CC OUTLET FCV.

C. 1SX007, U-1 SX OUTLET FROM CC HX VLV.

D. 1RH607, RH HX 1B FLOW CONT VLV.

Answer: C

Answer Explanation:

Question meets K/A, requires examinee knowledge of cause effect relationship between CC and SX systems.
The CC high temp alarm setpoint is 118°F, with a design limit of 120°F during the first 3 hours of RCS cooldown. Therefore the operator needs to take action to lower CC temp at the CC HX outlet.

Choice A is incorrect, throttle CC to FC HX is method used to adjust CC system pressure, NOT temperature. The affect would be lower overall CC system pressure and lower CC flow to the RH HX. This has a net result of raising CC HX outlet temp slightly when run on the Bwd simulator.
Choice B is incorrect, raising CC flow to the RH HX is not a method that would be used to control CC temperature. CC flow to the RH HX is set for proper design flow rates of the RH HX prior to CC system flow balancing being done. Raising the flow in this method would affect overall CC system pressure and possibly damage the RH HX.
Choice C is correct, BwAR 1-2-C5 immediate actions is to raise SX flow to the CC HX. This would have the largest impact on CC HX outlet temperature.
Choice D is incorrect, Raising RH flow through the RH HX would raise the heat transfer to the CC system and make the problem worse.

Question 7 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11061
User-Defined ID:	RS20008-N01
Cross Reference Number:	BWLI-CC1-062
Topic:	RS20008-N01
Num Field 1:	3.1
Num Field 2:	3.1
Text Field:	SYS008 K1.01
Comments:	Braidwood NRC Exam 2009 Question # 7 008 Component Cooling Water System (CCWS) Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: SWS TIER: 2 GROUP: 1 Cog Level: High Task No: R-CC-003 Obj No: 4C.CC-04 Cross Ref: 10CFR55.41(b)

General Data	
Technical Reference with Revision Number:	I1-CC-XL-01 rev.3a, ILT CC lesson plan
Justification for Non SRO CFR Link:	
Question History:	Not currently used in ILT program
Question Source:	ILT Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

8 **ID: RS20010-N01** **Points: 1.00**

Given:
- Unit 1 is in mode 5.
- BwOP RC-19, "VACUUM FILLING AN ISOLATED REACTOR COOLANT LOOP/PRESSURIZER" is in progress.
- Pressurizer level at 70% cold cal.
- Pressurizer pressure at 20" Hg vac.
- Vacuum pump is shut down and isolated.
- RH is in service with RCS loop temperatures at 115°F.
- Pressurizer heaters are energized.

Assuming pressurizer pressure remains constant, what will be the pressurizer liquid temperature when the bubble STARTS forming in the pressurizer?

A. 162°F

B. 193°F

C. 212°F

D. 228°F

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of steam table use in determining saturation temperature in pressurizer.
20"Hg is approx. 5 psia. which has saturation temp. of 162°F
Choice A is correct.
Choice B is incorrect, sat. temp for 10 psia. (common error for converting Hg to psia.)
Choice C is incorrect, sat. temp for atmospheric pressure.
Choice D is incorrect, sat. temp for 20 psia or 5 psig. (common error for converting Hg to psia.)

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:	11062
User-Defined ID:	RS20010-N01
Cross Reference Number:	22449
Topic:	RS20010-N01
Num Field 1:	3.1
Num Field 2:	3.4
Text Field:	SYS010 K5.01
Comments:	Braidwood NRC Exam 2009 Question # 8 010 Pressurizer Pressure Control System (PZR PCS) Knowledge of the operational implications of the following concepts as they apply to the RCS: Basic heat transfer concepts TIER: 2 GROUP: 1 Cog Level: High Task No: R-RC-038 Obj No: 3C.RC-14 Cross Ref: 10CFR55.41(b)

General Data	
Technical Reference with Revision Number:	BwOP RC-19 rev.15
Justification for Non SRO CFR Link:	
Question History:	Diablo Canyon 2002 NRCexam
Question Source:	Modified from INPO Bank #22449 (Diablo Canyon 02)
Low KA Justification (if required):	
Revision History:	Modified question with pressure that falls within BwOP RC-19 guidelines. Also changed pressure to inches of mercury vs. psia (vacuum pump gauge is in "Hg). Modified distractors to match common errors made in calculation.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

9 **ID: RS20012-N01** **Points: 1.00**

Given:
- Unit 2 is at 100% power, all systems normally aligned.
- An automatic reactor trip signal is generated.

- Due to a malfunction, SSPS Train 2A Logic Cabinet does NOT process the Reactor Protection logic coincidence.

- SSPS Train 2B functions as designed.

Based on the above conditions and assuming NO operator actions are taken, which ONE of the following would occur?

A. 2A Reactor Trip Breaker REMAINS CLOSED.
2B Reactor Trip Breaker OPENS with ONLY UV trip signals.

B. 2A Reactor Trip Breaker OPENS with ONLY UV trip signal.
2B Reactor Trip Breaker OPENS with UV AND shunt trip signals.

C. 2A Reactor Trip Breaker REMAINS CLOSED.
2B Reactor Trip Breaker OPENS with UV AND shunt trip signals.

D. 2A Reactor Trip Breaker OPENS with ONLY shunt trip signal.
2B Reactor Trip Breaker OPENS with UV AND shunt trip signals.

Answer: C

Answer Explanation:

Question meets K/A, requires examinee knowledge of logic circuit malfunction.
SSPS logic circuits provide train specific trips to the reactor trip breakers and opposite train trip to the reactor trip Bypass breakers. Therefore, failure of 2A train to process the logic coincidence would prevent 2A RTB from opening. 2B train would provide a trip signal to the 2A bypass breaker, but in a normally aligned condition the 2A Bypass breaker would be racked out.
Choice A is incorrect, RTB will receive trip from UV AND shunt.
Choice B is incorrect, RTA will not open.
Choice C is correct, see explanation above.
Choice D is incorrect, RTA will not open. Would be correct if 2A bypass breaker was racked in.

Question 9 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11063
User-Defined ID:	RS20012-N01
Cross Reference Number:	BWLI-RP1-029
Topic:	RS20012-N01
Num Field 1:	3.1
Num Field 2:	3.5
Text Field:	SYS012 K6.03
Comments:	Braidwood NRC Exam 2009 Question # 9 012 Reactor Protection System (RPS) Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Trip logic circuits TIER: 2 GROUP: 1 Cog Level: High Task No: R-RP-014 Obj No: 3S.RP-01 Cross Ref: 10CFR55.41(b)

General Data	
Technical Reference with Revision Number:	1-RP-XL-01 rev. 4, ILT SSPS lesson plan
Justification for Non SRO CFR Link:	
Question History:	Used in ILT systems
Question Source:	ILT Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

10 **ID: RS20012-N02** **Points: 1.00**

Given:
- Unit 2 is at 75% power.
- 24 hours ago, Pressurizer Pressure channel 2PT-457 failed.
- ALL actions of 2BwOA INST-2, OPERATION WITH FAILED INSTRUMENT CHANNEL have been completed.
- ALL associated Tech Spec required actions have been taken. (regardless of completion times)
- ALL other equipment (not associated with the 2PT-457 failure) is normally aligned.

- Today, with the above conditions, Instrument Bus 213 faults and de-energizes.

15 seconds after the instrument bus failure, with NO operator action, Unit 2 will...

A. have automatically tripped WITH safety injection actuated.

B. continue to operate at 75% power.

C. have automatically tripped, but with NO safety injection.

D. have automatically runback, until operator action is taken.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of bus power supplies to RPS channel.
Failure of 2PT-457 would require tripping bistables in Rx protection cabinet #3 for Hi Press. Rx Trip, Low Press. Rx Trip, Lo Press. SI, P11, OTDT Rx Trip, and OTDT Runback.
All of these coincidences are 2 of 4 (except P11 is 2 of 3). However, Inst. Bus 213 is also a power supply to protection cabinet 3, so a loss of Inst. Bus 213 would trip several bi-stables in the protection cabinet but not make up any coincidence for RPS or ESF actuation. The examinee needs to recognize that 2P-457 bistables and Inst. Bus 213 are both associated with RPS channel 3.

Choice A is incorrect, SI bistables do not trip from IB 213 loss and even if they did, they would be in same RPS channel so coincidence would not be met.
Choice B is correct, see explanation above
Choice C is incorrect, RX trip bistables do not trip from IB 213 loss and even if they did, they would be in same RPS channel so coincidence would not be met.
Choice D is incorrect, OTDT bistables do trip from IB 213 loss, however they are in same RPS channel so coincidence will not be met (bistable was already tripped).

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:	11089
User-Defined ID:	RS20012-N02
Cross Reference Number:	
Topic:	RS20012-N02
Num Field 1:	3.3
Num Field 2:	3.7
Text Field:	SYS012 K2.01
Comments:	Braidwood NRC Exam 2009 Question # 10 012 Reactor Protection System (RPS) Knowledge of bus power supplies to the following: RPS channels, components, and interconnections TIER: 2 GROUP: 1 Cog Level: High Task No: R-OA-006 Obj No: 3D.OA-22-A Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	ILT SSPS lesson plan I1-RP-XL-01 rev.4
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

11 **ID: RS20013-N01** **Points: 1.00**

Given:
- Unit 1 was at 100% power normal alignment.
- Subsequently, containment pressure transmitter 1PT934 has failed.
- All required actions of Tech Specs have been completed.

With the above conditions, the current coincidence logic for automatically actuating containment spray from OPERABLE pressure channels is...

A. 1 of 2.

B. 2 of 2.

C. 1 of 3.

D. 2 of 3.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of effect a detector malfunction will have on ESF actuation.
The CS actuation system has a total of four detectors with 2 of 4 logic. When one detector fails, tech spec 3.3.2 requires bypassing the failed CS bistable. This removes the failed channel from the CS logic circuit and the logic now becomes 2 of the remaining 3 detectors. The distractors are valid because 3 of the same detectors are used for SI and MSI signals. However, the tech spec actions and procedural requirements for those bistables are to trip the input rather than bypass it which would make the logic 1 of 3 for SI or MSI.
Choice A is incorrect, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is correct, see explanation above.

Question 11 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11066
User-Defined ID:	RS20013-N01
Cross Reference Number:	BWLI-CS1-044
Topic:	RS20013-N01
Num Field 1:	2.7
Num Field 2:	3.1
Text Field:	SYS013 K6.01
Comments:	Braidwood NRC Exam 2009 Question # 11 013 Engineered Safety Features Actuation System (ESFAS) Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: Sensors and detectors TIER: 2 GROUP: 1 Cog Level: High Task No: R-OA-100 Obj No: 3D.OA-10-J Cross Ref: 10CFR55.41(b)(7)

General Data	
Technical Reference with Revision Number:	I1-EF-XL-01 rev. 4b, ILT ESFAS lesson plan
Justification for Non SRO CFR Link:	
Question History:	Not currently selected for any ILT exams
Question Source:	ILT Bank
Low KA Justification (if required):	
Revision History:	revised stem and choices to clarify.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

12 **ID: RS20022-N01** **Points: 1.00**

Given:
- A reactor trip and safety injection have occurred on Unit 1.
- While performing step 7 of 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, an NSO reports the Group 2 RCFC Accident Mode lights for 1A and 1C RCFCs are dark.

Which of the following indications would explain the failure of the 1A and 1C RCFC Accident Mode lights to illuminate?

A. 1SX147A, 1A CNMT CHLR BYP ISOL VLV, OPEN light lit at 0PM02J.

B. 1SX027A, 1A/1C RCFC SX OUTLET VLV, OPEN light lit at 1PM06J.

C. 1SX016A, 1A/1C RCFC SX INLET VLV, OPEN light lit at 1PM06J.

D. 1SX112A, 1A CNMT CHLR INLET ISOL VLV, OPEN light lit at 0PM02J.

Answer: D

Answer Explanation:

The question meets the K/A/ requires examinee knowledge of logic used to assess status of containment cooling system in post accident conditions.
In order for the 1A/1C RCFC accident mode lights to light, all of the following must occur - 1A/1C RCFC low speed breaker closed,
1SX016A open, 1SX027A open, 1SX112A closed, 1SX114A closed, and 1SX147A open.
1SX112A open would prevent the 1A/1C RCFC accident mode light from illuminating.
All distractors are already in the correct position for accident mode lights to be lit.
Choice A is incorrect, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is correct, see explanation above.

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:	11067
User-Defined ID:	RS20022-N01
Cross Reference Number:	RS10022-008
Topic:	RS20022-N01
Num Field 1:	4.0
Num Field 2:	4.6
Text Field:	SYS022 2.4.21
Comments:	Braidwood NRC Exam 2009 Question # 12 022 Containment Cooling System (CCS) Knowledge of the parameters and logic used to assess the status of safety | functions, such as reactivity control, core cooling and heat removal, reactor | coolant system integrity, containment conditions, radioactivity release control, | etc. TIER: 2 GROUP: 1 Cog Level: Low Task No: R-VP-002 Obj No: 3C.VP-06-D Cross Ref: 10CFR55.41(b)(7)

General Data	
Technical Reference with Revision Number:	Big note VP-3 rev. 6 1BwEP-0 rev.200, step 7 RNO
Justification for Non SRO CFR Link:	
Question History:	2006 BWD cert Re-take exam
Question Source:	Modified from Bwd ILT Bank #RS10022-008
Low KA Justification (if required):	
Revision History:	Modified answer from 1SX147A closed to 1SX112A open and replaced distractor with 1SX147A open
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

13 **ID: RS20022-N02** **Points: 1.00**

Given:
- Unit 1 is at 100% power normal alignment.

- 1A SX pump is RUNNING.
- 1B SX pump is in STANDBY.

The following annunciator alarms at 1PM06J:
- 1-1-A2, CNMT LEAK DETECT FLOW HI
- An NSO reports that U-1 SX discharge pressure has dropped 5 psig since his MCB walkdown 15 minutes ago.
- 1PI-SX007, PP 1A DSCH PRESS is currently stable at 92 psig.

- No other alarms have come in and primary system parameters are all normal.
- The crew determines that a SX leak exists in U-1 containment and enter 1BwOA PRI-8, ESSENTIAL SERVICE WATER MALFUNCTION.

With the above conditions, per 1BwOA PRI-8, the FIRST MCR action(s) the crew will take to identify the leak location is...

A. shutdown 1A & C RCFCs and close 1SX016A and 1SX027A, RCFC 1A & C SX INLET and OUTLET VLVS.

B. shutdown the 1A SX pump and monitor RF sump pump run time.

C. shutdown 1B & D RCFCs and close 1SX016B and 1SX027B, RCFC 1B & D SX INLET and OUTLET VLVS.

D. verify 1SX016B and 1SX027B, RCFC 1A & C SX INLET and OUTLET VLVS are open and start the 1B SX pump.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of procedural mitigation strategy of large SX leak in Cnmt. cooling system.
With the above conditions the first actions in 1BwOA PRI-8 (step 4 RNO) are to shutdown and isolate the RCFC train for non-running (1B/D train) SX pump.
Choice A is incorrect because the 1SX016A/C valves will not close (interlocked) with the 1A SX pump running.
Choice B is incorrect. 1BwOA PRI-8 ensures at least one pump is running. At 100% power, the SX system has to remain running to supply vital loads outside containment and one train of containment cooling.
Choice C is correct, see explanation above.
Choice D is incorrect. Would be correct if the running pump had a low pressure alarm in but the question stem said the pressure dropped and stabilized above the low pressure alarm setpoint and no SX alarms have come in.

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:	11068
User-Defined ID:	RS20022-N02
Cross Reference Number:	
Topic:	RS20022-N02
Num Field 1:	3.1
Num Field 2:	3.5
Text Field:	SYS022 A2.05
Comments:	Braidwood NRC Exam 2009 Question # 13 022 Containment Cooling System (CCS) Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Major leak in CCS TIER: 2 GROUP: 1 Cog Level: High Task No: R-OA-108 Obj No: 3D.OA-69-D Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	1BwOA PRI-8 rev.102
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

14 **ID: RS20026-N01** **Points: 1.00**

Given:
- 10 minutes ago. a Unit 1 reactor trip and SI occurs due to a large steam break in containment.
- CS automatically actuated and all equipment operated as designed.
- NO operator actions have been taken on the CS system.
- The CS signal has NOT been reset.
- The crew is currently performing 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.
- The SM has directed the crew to isolate CS spray eductor flow by closing 1CS019A/B, EDUC SPRAY ADD VLVS, at the appropriate procedure step to minimize containment clean up.

The MINIMUM MCR action(s) required to isolate CS spray eductor flow is/are...

A. push CS reset pushbuttons and momentarily place 1CS019A/B control switches to close.

B. momentarily place 1CS019A/B control switches to close ONLY.

C. push CS reset pushbuttons ONLY (1CS019A/B will then auto close).

D. place CS Pump Test Switches to TEST, and momentarily place 1CS019A/B control switches to close.

Answer: A

Answer Explanation:

Question meets K/A, requires examinee knowledge of CS reset switches function and result of resetting CS. The 1CS019A/B valves auto open on a CS actuation signal. If the signal is not reset the valves will close with the C/S but will re-open immediately.
Choice A is correct, see explanation above.
Choice B is incorrect, CS signal is not reset.
Choice C is incorrect, 1CS019A/B will not auto close.
Choice D is incorrect, CS signal is not reset.

Question 14 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11069
User-Defined ID:	RS20026-N01
Cross Reference Number:	
Topic:	RS20026-N01
Num Field 1:	3.5
Num Field 2:	3.5
Text Field:	SYS026 A4.05
Comments:	Braidwood NRC Exam 2009 Question # 14 026 Containment Spray System (CSS) Ability to manually operate and/or monitor in the control room: Containment spray reset switches TIER: 2 GROUP: 1 Cog Level: Low Task No: R-CS-003 Obj No: 4D.CS-02 Cross Ref: 10CFR55.41(b)(7)

General Data	
Technical Reference with Revision Number:	1BwEP-1 rev. 200 step 7 Drwg. 20E-1-4030CS06 rev.L
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

15 **ID: RS20026-N02** **Points: 1.00**

Given:
- Unit 1 is 100% power normal alignment.
- Following maintenance on valve 1CS009A, PP 1A SUMP SUCT VLV, an NSO is performing 1BwOSR 5.5.8.CS-1A, TRAIN A CONTAINMENT SPRAY SYSTEM VALVE STROKE SURVEILLANCE.
- In order to meet the interlocks to open 1CS009A, EMD must install an electrical jumper to simulate which of the following valves is in the OPEN position?

A. 1RH8701A, RC LOOP 1A TO RH PP 1A SUCT ISOL VLV

B. 1SI8812A, PP 1A SUCT FROM RWST ISOL VLV

C. 1CS001A, PP 1A RWST SUCT VLV

D. 1SI8811A, CNMT SUMP 1A ISOL VLV

Answer: D

Answer Explanation:

Question meets the K/A, requires examinee knowledge of cause/effect relationship (interlock) between CS and ECCS system.
1SI8811A must be open prior to opening 1CS009A (ensures ECCS sump flow path to CS).
Distractors are all on the same 1A CS pump suction header but have different interlock or no interlock with 1CS009A.
Choice A is incorrect, 1RH8701A is interlocked to be closed prior to opening 1CS009A (ensures RCS hot leg not cross tied to CS suction)
Choice B is incorrect,1SI8812A has no interlock with 1CS009A but is on the same suction header
Choice C is incorrect,1CS009A must be open prior to opening 1CS001A (opposite of question).
Choice D is correct, see explanation above.

Question 15 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:	11070
User-Defined ID:	RS20026-N02
Cross Reference Number:	
Topic:	RS20026-N02
Num Field 1:	4.2
Num Field 2:	4.2
Text Field:	SYS026 K1.01
Comments:	Braidwood NRC Exam 2009 Question # 15 026 Containment Spray System (CSS) Knowledge of the physical connections and/or cause effect relationships between the CSS and the following: ECCS TIER: 2 GROUP: 1 Cog Level: Low Task No: R-AM-017 Obj No: 3E.AM-023-A Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	1BwOSR 5.5.8.CS-1A rev.10
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

16 **ID: RS20039-N01** **Points: 1.00**

Given:
- Unit 1 is in Mode 3 following a normal reactor shutdown.
- A cooldown commenced at 1500 following 2 hours of stable RCS temperature.

The following data has been recorded on 1BwOSR 3.4.3.1, RCS PRESSURE/TEMPERATURE LIMIT SURVEILLANCE.

Time	Loop A Cold Leg Temp	Temp (°F) Change	Rate (°F/hr)	Temp Change in 1 hr Period
1500	557°F	0	0	0
1515	525°F	32	128	32
1530	499°F	26	104	58
1545	479°F	20	80	78
1600	460°F	19	76	97
1615	430°F	30	120	95
1630	395°F	35	140	104
1645	380°F	15	60	99
1700	374°F	16	64	86

The RCS cooldown FIRST exceeded the limits of Tech Spec. 3.4.3, RCS P/T LIMITS, at _____, which may have exceeded the assumed _____ .

A. 1515, margin to RCS brittle failure.

B. 1630, shutdown margin.

C. 1515, shutdown margin.

D. 1630, margin to RCS brittle failure.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee to know how cooldown rates are calculated (operational implication) and what the consequence is when exceeded.
The maximum cooldown rate as defined in the PTLR is 100°F in any 1 hr period.
1BwOSR 3.4.3.1 directs that rate to be calculated at 30 min intervals even though data can be taken more often for trending. At 1515 the rate exceeded 100°F , but it was not in a 1 hr. period. The 1630 calculation was the first 1 hr. period in which the 100°F limit was violated. The bases for the cooldown rate is to establish limits to provide a margin to brittle failure of the RCS and vessel.
Choice A is incorrect, wrong time.
Choice B is incorrect, due to wrong reason.Shutdown margin is valid distractor because the bases for adequate SDM assumes a severe RCS temperature drop during a MSLB event and positive reactivity due to MTC. This can easily be confused with rate of temperature drop.
Choice C is incorrect, wrong time and reason.
Choice D is correct, see explanation above.

Question 16 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:	11071
User-Defined ID:	RS20039-N01
Cross Reference Number:	
Topic:	RS20039-N01
Num Field 1:	2.7
Num Field 2:	3.1
Text Field:	SYS039 K5.05
Comments:	Braidwood NRC Exam 2009 Question # 16 039 Main and Reheat Steam System (MRSS) Knowledge of the operational implications of the following concepts as the apply to the MRSS: Bases for RCS cooldown limits TIER: 2 GROUP: 1 Cog Level: High Task No: R-MS-002 Obj No: 3C.MS-06-A Cross Ref: 10CFR55.41(b) (5)

General Data	
Technical Reference with Revision Number:	TS 3.4.3 1BwOSR 3.4.3.1 Rev.4
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

17 **ID: RS20059-N01** **Points: 1.00**

Given:
- Unit 1 is at 80% power, normal alignment.
- 1A Main Feedwater Pump is OOS.
- 1B Main Feedwater Pump is running.
- 1C Main Feedwater Pump just tripped.

The NSO closes 1FW012C, RECIRC VALVE, then attempts to runback the turbine with both the Runback pushbutton and OWS graphic 5512 poke field. However, the turbine load does NOT change.

The NSO should then ...

A. program DEHC and initiate a 20 MW/min load drop to 780 MW.

B. manually trip the reactor and enter 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION.

C. start the S/U Feedwater Pump.

D. program DEHC and initiate a 250 MW/min load drop to 700 MW.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of power restrictions of MFW pumps.
Per 1BwOA SEC-1 the turbine is runback to 700 MW at 250 MW per min with only one FW pump operating.
Choice A is incorrect, would be correct if only one HD pump was available.
Choice B is incorrect, tripping the reactor may be necessary, however only if a manual runback cannot be executed.
Choice C is incorrect, there is no action in 1BwOA SEC-1 for starting the SU FW pump.
Choice D is correct, see explanation above.

Question 17 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11073
User-Defined ID:	RS20059-N01
Cross Reference Number:	BWLI-SEC1019
Topic:	RS20059-N01
Num Field 1:	2.7
Num Field 2:	2.9
Text Field:	SYS059 A1.03
Comments:	Braidwood NRC Exam 2009 Question # 17 059 Main Feedwater (MFW) System Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Power level restrictions for operation of MFW pumps and valves TIER: 2 GROUP: 1 Cog Level: Low Task No: R-OA-004 Obj No: 4D.OA-20 Cross Ref: 10CFR55.41(b) (4)

General Data	
Technical Reference with Revision Number:	1BwOA SEC-1 rev.104
Justification for Non SRO CFR Link:	
Question History:	Not currently used in ILT
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	modified a distractor.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

18 **ID: RS20061-N01** **Points: 1.00**

Given:
- A Unit 1 RCS cooldown is in progress with the 1A AF pump maintaining stable level in all steam generators.
- Then instrument power is lost to the 1AF005A controller.

Assuming RCS heat input and SG steaming rate remain constant, 1AF005A, SG 1A FLOW CONT VLV ...

A. fails closed and 1A SG level will drop.

B. fails open and 1A SG level will rise.

C. fails as is and 1A SG level will remain the same.

D. is NOT affected and 1A SG level will remain the same.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of failure in AF system affect on SG level.
If inst. power is lost to the 1AF005A-D valve controllers, zero flow is demanded and the valves throttle closed to achieve zero flow.
Choice A is correct, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above but plausible because power loss was AC vs. DC (DC loss would have no affect).

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:	11074
User-Defined ID:	RS20061-N01
Cross Reference Number:	
Topic:	RS20061-N01
Num Field 1:	4.2
Num Field 2:	4.4
Text Field:	SYS061 K3.02
Comments:	Braidwood NRC Exam 2009 Question # 18 061 Auxiliary / Emergency Feedwater (AFW) System Knowledge of the effect that a loss or malfunction of the AFW will have on the following: S/G TIER: 2 GROUP: 1 Cog Level: Low Task No: R-OA-006 Obj No: 3D.OA-22-A Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	I1-AF-XL-01 rev. 4b Bwd ILT Aux. Feedwater Lesson Plan
Justification for Non SRO CFR Link:	
Question History:	? (INPO bank says Bwd 2000 exam but I cannot find it on our 2000 exam)
Question Source:	INPO Bank
Low KA Justification (if required):	
Revision History:	revised question to include SG level response in answers to better fit K/A
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

19 **ID: RS20062-N01** **Points: 1.00**

Given:
- The MCR is receiving sporadic alarms from annunciator 1-4-D5, BUS 114 INVERTER TROUBLE.
- Plant engineering suspects an intermittent ground on Instrument Bus 114, and is making preparations for trouble shooting.
- The Unit Supervisor asks for Instrument Bus 114 current to be monitored for fluctuations.

Instrument Bus 114 amps can best be monitored at...

A. Instrument Bus 114, in the U-1 Aux Electrical Equipment Room.

B. Instrument Inverter 114 in the U-1 Misc. Electrical Equipment Room.

C. Division 12 section of 1PM01J, in the Main Control Room.

D. N44 drawer of 1PM07J, in the Main Control Room.

Answer: B

Answer Explanation:

Question meets K/A, requires examinee knowledge of where to monitor vital AC bus amperage.
Choice A is incorrect, The AC instrument buses do not have any local amperage meters.
Choice B is correct, the instrument inverter has inverter output amp meter. Also annunciator response directs EO dispatched to instrument inverter.
Choice C is incorrect, their is no ESF bus amp meter smaller than 480V bus on 1PM01J.
Choice D is incorrect, the N-44 drawers, although fed from instrument bus 114, only have meters for detector current, not bus current.

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:	11075
User-Defined ID:	RS20062-N01
Cross Reference Number:	
Topic:	RS20062-N01
Num Field 1:	3.0
Num Field 2:	3.1
Text Field:	SYS062 A3.01
Comments:	Braidwood NRC Exam 2009 Question # 19 062 AC Electrical Distribution System Ability to monitor automatic operation of the ac distribution system, including: Vital ac bus amperage TIER: 2 GROUP: 1 Cog Level: Low Task No: R-AP-001 Obj No: 4C.AP-01 Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	Big note I&C-5 rev.6 BwAR 1-4-A5 rev.52
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

20 **ID: RS20063-N01** **Points: 1.00**

Given:
- Unit 1 is in mode 5, with 1BwGP 100-1, PLANT HEATUP, in progress.
- The 1A CV pump has just been started.
- Five seconds AFTER the pump start, DC control power is LOST to bus 141 due to a faulty fuse.
- NO other plant components are damaged.

After the loss of control power the 1A CV pump...

A. can ONLY be tripped locally by mechanically actuating the trip coil.

B. will trip immediately from a loss of DC control power.

C. can be tripped locally with the breaker local control switch.

D. can be tripped remotely from the MCR switch.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of DC electrical system design features that provide for trips. 4KV pump control circuits have trip coils that are energize to actuate. Therefore, when DC control power is lost to a running 4KV pump, the only way to trip the pump is locally by mechanical actuation of the trip coil.
Choice A is correct, see explanation above.
Choice B is incorrect, trip coils are energize to actuate.
Choice C is incorrect, local control switch needs power to energize trip coil.
Choice D is incorrect, MCR control switch needs power to energize trip coil.

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:	11077
User-Defined ID:	RS20063-N01
Cross Reference Number:	
Topic:	RS20063-N01
Num Field 1:	2.6
Num Field 2:	2.9
Text Field:	SYS063 K4.04
Comments:	Braidwood NRC Exam 2009 Question # 20 063 DC Electrical Distribution System Knowledge of DC electrical system design feature(s) and/ or interlock(s) which provide for the following: Trips TIER: 2 GROUP: 1 Cog Level: Low Task No: R-DC-003 Obj No: 4C.DC-01 Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	PWR Generic Fundamentals Components lesson plan, rev. 2
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

21 **ID: RS20064-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power.
- The 1A Diesel Generator has auto started and energized bus 141 due to a undervoltage condition on the bus.
- Equipment repairs are complete and operators are currently performing BwOP AP-32, SYNCHRONIZING A SAT TO A BUS BEING FED BY A DG, to re-align bus 141 feed to normal.

Currently:
- DG 1A EMERG MODE SPEED/VOLT CONTROL switch is in MAN EMERG MODE.
- The normal feed breaker to bus 141 is closed with the DG and SAT in parallel.

With the above conditions, if the NSO places DG 1A EMERG MODE SPEED/VOLT CONTROL switch back to AUTO, initially the 1A DG...

A. engine will trip on reverse power.

B. will pick up real load (WATTS) from the grid.

C. will speed up to the mechanical governor set point.

D. will reject reactive load (VARS) to the grid.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of system precaution and explanation for precaution. BwOP AP-32, precaution D.4. cautions the operator not to return the EMERG MODE SPEED/VOLT CONTROL switch to auto while the DG is parallel to the grid to prevent the DG from going to isochronous mode. Returning the DG to isochronous mode while parallel to the grid will cause DG to try to raise speed to mechanical governor setpoint (approx. 60.8 Hz). However, since the DG is still connected to the grid, speed will not change, thereby causing real load to rise excessively.
Choice A is incorrect, reverse power trip for the engine is not active during auto start of DG.
Choice B is correct, see explanation above.
Choice C is incorrect, the DG will try to speed up, but will not be able top since it is still connected to the grid.
Choice D is incorrect, reactive load is affected by excitation which will not change.

Question 21 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11078
User-Defined ID:	RS20064-N01
Cross Reference Number:	
Topic:	RS20064-N01
Num Field 1:	3.8
Num Field 2:	4.0
Text Field:	SYS064 2.1.32
Comments:	Braidwood NRC Exam 2009 Question # 21 064 Emergency Diesel Generator (ED/G) System Ability to explain and apply system limits and precautions. TIER: 2 GROUP: 1 Cog Level: High Task No: R-DG-015 Obj No: 3C.DG-02-B Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	BwOP AP-32 rev. 4 Big Note DG-3 rev.8
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

22 **ID: RS20064-N02** **Points: 1.00**

Which of the following combinations of Unit 1 breaker positions will initiate the division 11 Blackout Sequencer?

	1412	**1413**	**1414**
A.	CLOSED	OPEN	CLOSED
B.	OPEN	CLOSED	CLOSED
C.	OPEN	CLOSED	OPEN
D.	CLOSED	OPEN	OPEN

A. A

B. B

C. C

D. D

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of interlocks which provide for the automatic blackout sequencer.
Blackout sequencer will start when DG output breaker (1413) is closed with BOTH bus 141 feed breakers open (1412 and 1414).

Choice A is incorrect, all three breakers positions incorrect.
Choice B is incorrect, 1414 breaker position incorrect.
Choice C is correct, see explanation above.
Choice D is incorrect, 1412 and 1414 breaker positions incorrect.

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:	11079
User-Defined ID:	RS20064-N02
Cross Reference Number:	RS10064-004
Topic:	RS20064-N02
Num Field 1:	3.5
Num Field 2:	4.0
Text Field:	SYS064 K4.10
Comments:	Braidwood NRC Exam 2009 Question # 22 064 Emergency Diesel Generator (ED/G) System Knowledge of ED/G system design feature(s) and/or interlock(s) which provide for the following: Automatic load sequencer: blackout TIER: 2 GROUP: 1 Cog Level: Low Task No: R-DG-003 Obj No: 3C.DG-01-B Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	P&ID 20E-1-4030EF01 Big Note DG-2 rev.7
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	

23 **ID: RS20073-N01** **Points: 1.00**

Given:
- Communications loop failures occur on ALL RM-11 grid 2 radiation monitors.

With the above condition, ALL of the following listed grid 2 monitors can be monitored from a Main Control Room RM-23, EXCEPT...

A. 1PR11J, CNMT ATMOSPHERE

B. 1PR30J, VENT STACK 0A EFFLUENT

C. 0PR35J, MCR TURB BLDG AIR INLET

D. 0PR31J, MCR OUTSIDE AIR INLET

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of where rad levels can be monitored from process radiation skids.
Normally, all rad skids rad levels are monitored from the RM-11 computer. If a communication loop failure occurs the RM-11 will lose indication for the affected monitors. The three incorrect choices each have an RM-23 (alternate indication) in the MCR that could be used to monitor rad levels during a communications loop failure. The 1PR011J does not have an RM-23, therefore it would lose all MCR indication.

Choice A is correct, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

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:	11080
User-Defined ID:	RS20073-N01
Cross Reference Number:	
Topic:	RS20073-N01
Num Field 1:	3.2
Num Field 2:	3.5
Text Field:	SYS073 A1.01
Comments:	Braidwood NRC Exam 2009 Question # 23 073 Process Radiation Monitoring (PRM) System Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including: Radiation levels TIER: 2 GROUP: 1 Cog Level: Low Task No: R-AR-002 Obj No: 4C.AR-02 Cross Ref: 10CFR55.41(b) (11)

General Data	
Technical Reference with Revision Number:	Big Note AR-1 rev.8 ILT lesson plan I1-AR-XL-01 rev. 4a
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

24 **ID: RS20076-N01** **Points: 1.00**

Given:
- Both units are at 100% power.
- 1A Circ Water Pump forebay is drained for silt removal.
- 0SX115A, UPST ISOL TO SX PUMPS TRAIN A SUCTION HDR FROM 1A BAY, is danger tagged CLOSED.
- All other SX system equipment is in normal alignment.

An event occurs that causes debris in the 1B Circ Water Pump forebay to block flow at the inlet to the SX suction piping.
ONLY the 1B Circ Water Pump forebay is affected.

If ALL SX pumps were running with the above conditions, the control room NSO would likely see cavitation from the _____ SX Pump(s).

A. 1A AND 2A

B. 1A (ONLY)

C. 1A AND 1B

D. 1B AND 2B

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee to predict the impacts of a potential loss of service water. Question only test part (a) of K/A IAW NUREG ES-401 section D.2.a (page 6) because part (b) does not create a discriminating question.
The SX suction header is designed such that the 3 U-1 forebays each combine to a common suction header to the 1A and 2A SX pumps. Likewise the U-2 forebays combine to the "B" train suction header. The system is designed such that it can operate full capacity with any 1 of the 3 forebays out of service and the respective 0SX115 valve closed. If debris were to block a second forebay on the unit 1, it would affect the train A SX pumps on both units.

Choice A is correct, see explanation above
Choice B is incorrect, distractor is valid if examinee is unaware that U-1 forebays also supply U-2 SX pump.
Choice C is incorrect, distractor is valid if examinee assumes that U-1 forebays supply both U-1 SX pumps.
Choice D is incorrect, distractor is valid if examinee assumes that 1B forebay supplies 1B and 2B SX pumps.

Question 24 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:	11081
User-Defined ID:	RS20076-N01
Cross Reference Number:	
Topic:	RS20076-N01
Num Field 1:	3.5
Num Field 2:	3.7
Text Field:	SYS076 A2.01
Comments:	Braidwood NRC Exam 2009 Question # 24 076 Service Water System (SWS) 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: Loss of SWS TIER: 2 GROUP: 1 Cog Level: High Task No: R-SX-006 Obj No: 4C.SX-03 Cross Ref: 10CFR55.41(b) (4)

General Data	
Technical Reference with Revision Number:	Big Note SX-1 rev.15 Bwd ILT lesson plan I!-SX-XL-01 rev. 4b
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

25 **ID: RS20062-N02** **Points: 1.00**

Given:
- A reactor trip and SI have occurred on Unit 1.
- The crew is currently performing 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, at step 3, VERIFY POWER TO 4 KV ESF BUSES.

In the above situation,
(1) the condition for which 1BwOA ELEC-3, LOSS OF 4KV ESF BUS, applies is...
(2) the actual application of 1BwOA ELEC-3 is...

A. (1) a loss of ONLY one ESF bus.
(2) perform immediately (while continuing with 1BwEP-0).

B. (1) a loss of ONLY one ESF bus.
(2) perform when time permits (while continuing with 1BwEP-0).

C. (1) a loss of ONLY one ESF bus.
(2) transition to (suspend usage of 1BwEP-0).

D. (1) a loss of BOTH ESF buses
(2) transition to (suspend usage of 1BwEP-0).

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of how abnormal operating procedure (1BwOA ELEC-3) is used in conjunction with EOPs (1BwEP-0)
1BwOA ELEC-3 is used with only 1 ESF bus de-energized in step 3.b of 1BwEP-0. (At least one ESF bus has to be energized in step 3.a in order to get to step 3.b). If both buses are de-energized the procedure transition would be to 1BwCA 0-0 in step 3.a.
Since step 3 is an immediate action step, the word usage "When time permits" is required knowledge for a RO.
Choice A is incorrect, part (2) is "When time permits"
Choice B is correct, see explanation above.
Choice C is incorrect, part (2) is "When time permits"
Choice D is incorrect, transition would be to 1BwCA 0.0 with both ESF buses de-energized.

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:	11082
User-Defined ID:	RS20062-N02
Cross Reference Number:	
Topic:	RS20062-N02
Num Field 1:	3.8
Num Field 2:	4.5
Text Field:	SYS062 2.4.08
Comments:	Braidwood NRC Exam 2009 Question # 25 062 AC Electrical Distribution System Knowledge of how abnormal operating procedures are used in conjunction with | EOPs. TIER: 2 GROUP: 1 Cog Level: Low Task No: R-OA-131 Obj No: 3D.OA-34-C Cross Ref: 10CFR55.41(b)(10)

General Data	
Technical Reference with Revision Number:	1BwEP-0 rev. 200
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

26 **ID: RS20078-N01** **Points: 1.00**

Given:
- Both units are at 100%, normal alignment.
- Unit 1 SAC and IA drier are on line.

An event occurs and the MCR receives the following alarms:
- 0-37-C3, IA RCVR 1 PRESS LOW.
- 0-37-D3, IA DRYER 1 TROUBLE (15 sec. later).

NO other alarms come in.

The NSO notes the following:
- Station Air pressure is 115 psig and stable.
- Instrument Air pressure dropped to 80 psig and is slowly rising.

Based on the above indications, which of the following is the most likely event that occured?

A. An instrument air system leak caused a stand-by air compressor to start.

B. The U-1 IA Receiver relief valve momentarily lifted and reclosed.

C. A U-1 IA dryer purge valve failed open and the air dryer bypass valve automatically opened.

D. The running station air compressor tripped and caused a stand-by compressor to start.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of interlocks that cross tie Instrument air to other air systems (station air).
A failed open purge valve on the air dryer will cause IA pressure to drop until IA receiver low press alarms at 94 psig. The air dryer bypass valve auto opens at 80 psig.

Choice A is incorrect, stand by compressor starts on a low station air receiver pressure (not IA receiver) and stand by compressor start would cause additional alarms.
Choice B is incorrect, receiver relief lifting would not cause IA dryer trouble alarm unless system pressure dropped to 50 psig. Stem has pressure only dropping to 80 psig.
Choice C is correct, see explanation above.
Choice D is incorrect, compressor trip and and stand by compressor start would cause additional alarms.

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:	11083
User-Defined ID:	RS20078-N01
Cross Reference Number:	
Topic:	RS20078-N01
Num Field 1:	3.2
Num Field 2:	3.5
Text Field:	SYS078 K4.02
Comments:	Braidwood NRC Exam 2009 Question # 26 078 Instrument Air System (IAS) Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Cross-over to other air systems TIER: 2 GROUP: 1 Cog Level: High Task No: R-IA-002 Obj No: ? Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	BwAR 0-37-C3 rev. 6 BwAR 0-37-D3 rev.7
Justification for Non SRO CFR Link:	
Question History:	?
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

27 **ID: RS20078-N02** **Points: 1.00**

Given:
- Both units are at 100% power, normal alignment.

An air system leak in the plant causes the following alarms in the MCR:
- 0-37-C2, SAC 1 RCVR PRESS LOW.
- 0-37-C3, IA RCVR 1 PRESS LOW.

- The U-1 MCR crew enters 1BwOA SEC-4, LOSS OF INSTRUMENT AIR.
- The US instructs an extra NSO to monitor for reactor trip criteria per 1BwOA SEC-4.

The correct meter the NSO should monitor is...

A. 0PI-IA007, INST AIR HDR PRESS at 0PM01J

B. 0PI-SA006, SERV AIR HDR PRESS at 0PM01J

C. 1PI-GS001, SUP HDR PRESS at 1PM02J

D. 1PI-ES043, EXH HOOD 1A PRESS at 1PM03J

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to monitor pressure gauges in the MCR.

Choice A is correct 1BwOA SEC-4, step 3 checks IA press at 0PI-IA007 (checking for adequate IA supply to operate FW valves) < 60 psig, if yes, next step is to trip reactor.
Choice B is incorrect, pressure transmitter for 0PI-SA006 is on the main turb. bldg SA header. The IA supply taps off the compressor discharges upstream of the SA header. depending on the location and size of the leak, the 0PI-SA006 may be less than 0PI-IA007.
Choice C is incorrect, 1BwOA SEC-4 checks GS supply pressure at step 9, but no reactor trip criteria is mentioned.
Choice D is incorrect, this gauge would be correct for a loss of condenser vacuum, but not inst.air.

Question 27 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:	11084
User-Defined ID:	RS20078-N02
Cross Reference Number:	
Topic:	RS20078-N02
Num Field 1:	3.1
Num Field 2:	3.1
Text Field:	SYS078 A4.01
Comments:	Braidwood NRC Exam 2009 Question # 27 078 Instrument Air System (IAS) Ability to manually operate and/or monitor in the control room: Pressure gauges TIER: 2 GROUP: 1 Cog Level: Low Task No: R-IA-002 Obj No: 4C.IA-01 Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	1BwOA SEC-4 rev. 101
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

28 **ID: RS20103-N01** **Points: 1.00**

Given:
- Unit 2 is 100% power.

- 2C RCFC HIGH AND LOW speed fans are danger tagged OOS.
- All other equipment is normally aligned.

While performing 2BwOSR 0.1-1,2,3, SHIFTLY DAILY OPERATING SURV, the NSO observes the following RCFC temperature indications on 2PM06J:

- 2A INLET TEMP - 95°F
- 2A OUTLET TEMP - 70°F

- 2C INLET TEMP - 105°F
- 2C OUTLET TEMP - 75°F

- 2B INLET TEMP - 95°F
- 2B OUTLET TEMP - 70°F

- 2D INLET TEMP - 95°F
- 2D OUTLET TEMP - 70°F

With the above indications, which of the following is the containment average air temperature that should be recorded for Tech. Spec. 3.6.5. CONTAINMENT AIR TEMPERATURE surv. requirement?

A. 82.5 °F

B. 84.4 °F

C. 95.0°F

D. 97.5 °F

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee ability monitor containment temperature to prevent exceeding design limit (tech spec)
The method for determining containment average air temperature for tech spec 3.6.5 is to average the inlet temps of the operating RCFCs. Since 2C RCFC is shut down, its inlet temp is not used in the average calculation.
Choice A is incorrect, this is average of 2A,B & D inlet AND outlet temps. Outlet temps should not be used.
Choice B is incorrect, this is average of ALL inlet AND outlet temps. Outlet temps should not be used and 2C inlet temp should not be used.
Choice C is incorrect, see explanation above.
Choice D is incorrect, this is average of ALL inlet temps. 2C temps should not be used.

Question 28 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11085
User-Defined ID:	RS20103-N01
Cross Reference Number:	
Topic:	RS20103-N01
Num Field 1:	3.7
Num Field 2:	4.0
Text Field:	SYS103 A1.01
Comments:	Braidwood NRC Exam 2009 Question # 28 103 Containment System Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: Containment pressure, temperature, and humidity TIER: 2 GROUP: 1 Cog Level: High Task No: R-VP-002 Obj No: 3C.VP-06-B Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	2BwOSR 0.1-1,2,3 rev. 53
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

29 **ID: RS20002-N01** **Points: 1.00**

Given:
- An RCS cooldown and depressurization is in progress following an RCS leak.

Steady state plant conditions have been established.
- CETCs are 532 °F.
- RCS pressure is 885 psig and STABLE.
- S/G pressures are 910 psig.
- Steam dumps are 5% open.
- AF flow is 20 gpm to each S/G.

Based on these conditions, the operator can aid in establishing subcooling by LOWERING...

A. RCS pressure.

B. Steam Dump demand.

C. Aux. Feedwater flows.

D. S/G pressures.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of
operational implications as they apply to saturation/subcooling conditions.
Choice A is incorrect, lower RCS pressure would make subcooling worse, not better.
Choice B is incorrect, lower steam dump demand would remove less heat from the RCS and raise RCS temp making subcooling worse, not better.
Choice C is incorrect, lower AF flow would remove less heat from the RCS and raise RCS temp making subcooling worse, not better.
Choice D is correct lowering SG press would remove more heat from the RCS and improve subcooling.

Question 29 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11086
User-Defined ID:	RS20002-N01
Cross Reference Number:	BWLC3DEP2005
Topic:	RS20002-N01
Num Field 1:	3.7
Num Field 2:	4.2
Text Field:	SYS002 K5.09
Comments:	Braidwood NRC Exam 2009 Question # 29 002 Reactor Coolant System (RCS) Knowledge of the operational implications of the following concepts as they apply to the RCS: Relationship of pressure and temperature for water at saturation and subcooling conditions TIER: 2 GROUP: 2 Cog Level: High Task No: R-SG-001 Obj No: 3S.MS-01 Cross Ref: 10CFR55.41(b) (5)

General Data	
Technical Reference with Revision Number:	None
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

30 **ID: RS20011-N01** **Points: 1.00**

Given:

- Unit 1 is at 50% power.
- Pressurizer pressure is 2240 psig.
- Tave = Tref.
- 1FK-121 controller, CENT CHG PMPS FLOW CONT VLV is controlling charging flow in MANUAL.
- Pressurizer level is 50%.
- ALL other controls are in AUTOMATIC.
- 1A and 1B backup heater groups have automatically energized.
- 1D backup heater group is NOT energized.

With the above conditions, the backup heaters groups are ...

A. responding properly.

B. NOT responding properly, NO groups should be ENERGIZED.

C. NOT responding properly, ONLY the 1A group should be ENERGIZED.

D. NOT responding properly, ALL groups should be ENERGIZED.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of effect of malfunction of relationship between Pzr level and heater control circuit.
42.5% is the programmed PZR level for 50% power (Tave of 572°F), +5% above program is the level which energizes ALL the backup heaters. All back up heaters should work in unison in automatic. When back up heaters are controlled in manual they can be selected for one or two groups to energize independently.

Choice A is incorrect, all groups should be on.
Choice B is incorrect, all groups should be on.
Choice C is incorrect, all groups should be on.
Choice D is correct, see explanation above.

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:	11087
User-Defined ID:	RS20011-N01
Cross Reference Number:	RS20011-004
Topic:	RS20011-N01
Num Field 1:	2.9
Num Field 2:	3.3
Text Field:	SYS011 K6.03
Comments:	Braidwood NRC Exam 2009 Question # 30 011 Pressurizer Level Control System (PZR LCS) Knowledge of the effect of a loss or malfunction on the following will have on the PZR LCS: Relationship between PZR level and PZR heater control circuit TIER: 2 GROUP: 2 Cog Level: High Task No: R-RY-015 Obj No: 4C.RY-09 Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	BwAR 1-12-C3 rev. 52
Justification for Non SRO CFR Link:	
Question History:	?
Question Source:	Modified from Bwd ILT bank #RS20011-004
Low KA Justification (if required):	
Revision History:	Changed power level in stem from 60% to 50%. Changed question subject from calculation of pressurizer level when htrs energize to recognition of proper htr operation.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

31 **ID: RS20014-N01** **Points: 1.00**

Given:
- 5 minutes ago, Unit 1 reactor was manually tripped from 5% power during a normal shutdown.
- All systems operated as designed.
- Current SG levels are ALL 50% and stable.
- ALL alarms at 1PM05J have been acknowledged, but NOT reset.

With the above conditions, if the NSO momentarily places the ROD CONT STARTUP switch on 1PM05J, to the START UP position ...

A. all control rod group step counters will reset to zero.

B. alarm 1-10-E6, ROD AT BOTTOM will clear (slow flash).

C. the reactor trip breakers will close.

D. alarm 1-11-A1, MANUAL RX TRIP will clear (slow flash).

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to monitor result of reset switch used for re-zeroing rod position in MCR.
The ROD CONT STARTUP switch re-zero's the following:
P/A converter, slave cyclers, step counters, rod control internal memory, and master cycler. Only step counters can be monitored from the MCR.

Choice A is correct, see explanation above.
Choice B is incorrect, Rod At Bottom alarm will not clear until control bank A is withdrawn during a start up.
Choice C is incorrect, reactor trip breakers have separate switch to close on 1PM05J. Distractor is plausible because with SG levels above lo-2 setpoint, there should not be an active Rx trip signal.
Choice D is incorrect, First out alarm has separate keyed reset switch on 1PM05J.

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:	11088
User-Defined ID:	RS20014-N01
Cross Reference Number:	
Topic:	RS20014-N01
Num Field 1:	2.7
Num Field 2:	2.7
Text Field:	SYS014 A4.04
Comments:	Braidwood NRC Exam 2009 Question # 31 014 Rod Position Indication System (RPIS) Ability to manually operate and/or monitor in the control room: Re-zeroing of rod position prior to startup TIER: 2 GROUP: 2 Cog Level: low Task No: R-RD-014 Obj No: 3C.RD-02-A Cross Ref: 10CFR55.41(b) (6)

General Data	
Technical Reference with Revision Number:	I1-RD-XL-01 rev. 3a Bwd ILT Rod Control lesson plan
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

32 **ID: RS20015-N01** **Points: 1.00**

Given:
- Unit 1 reactor is critical at $1x10^{-8}$ amps (BOTH IR channels).
Then:
- Instrument Bus 111 is deenergized.

With the above condition and NO operator action, the reactor will trip on high flux due to the loss of power to _____.

A. Permissive P-10

B. SR channel N-31

C. IR channel N-35

D. PR channel N-41

Answer: C

Answer Explanation:

Question meets K/A, requires examinee to know bus power supplies to NIS channels and components.
Bus 111 is power supply to SR N31 and IR N35 and PR N41. At $1x10^{-8}$ amps power is above P-6 and below P10. The SR instruments would already be de-energized and PR high flux trips are 2 of 4 coincidence, therefore the rx trip will come from IR high flux. Specifically, due to a loss of power to N35. Bus power loss would not affect P-10 since it is a 2 of 4 coincidence also.
Choice A is incorrect, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is correct, see explanation above.
Choice D is incorrect, see explanation above.

Question 32 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:	11064
User-Defined ID:	RS20015-N01
Cross Reference Number:	BWLC3DOA2003
Topic:	RS20015-N01
Num Field 1:	3.3
Num Field 2:	3.7
Text Field:	SYS015 K2.01
Comments:	Braidwood NRC Exam 2009 Question # 32 015 Nuclear Instrumentation System (NIS) Knowledge of bus power supplies to the following: NIS channels, components, and interconnections TIER: 2 GROUP: 1 Cog Level: High Task No: R-NI-003 Obj No: 3C.NI-04-C Cross Ref: 10CFR55.41(b)

General Data	
Technical Reference with Revision Number:	Bwd Big note NI-3 rev.9
Justification for Non SRO CFR Link:	
Question History:	?
Question Source:	Bwd LORT Bank
Low KA Justification (if required):	
Revision History:	Revised to change from 1 of 2 taken twice format.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

33 **ID: RS20016-N01** **Points: 1.00**

Given:
- Unit 1 is preparing to enter MODE 4 from MODE 3.
- RCS pressure is being controlled at 650 psig.
- ALL Wide Range RCS temperatures are 350°F.
- BOTH PZR PORV control switches are in the "ARM LOW TEMP" position at 1PM05J.

With the above conditions, if 1TE413E, Loop 1B Tcold Wide Range temperature sensor, fails LOW, _____ will automatically OPEN.

(U1 PTLR Figure 3.1 is attached)

A. ONLY PZR PORV, 1RY455A

B. ONLY PZR PORV, 1RY456

C. BOTH PZR PORVs, 1RY455A and 1RY456

D. NEITHER PZR PORV, 1RY455A nor 1RY456

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to monitor automatically selecting (auctioneering) inputs to control function.
The LTOP control circuits use auctioneered low WR Thot for Pzr PORV 1RY456 and auctioneered low WR Tcold for Pzr PORV 1RY455A. At 650 psig a failed low auct. temp input would cause either PORV to reach the lift setpoint, however the stem specifies a Tcold failure which would only affect 1RY456.

Choice A is incorrect, see explanation above.
Choice B is correct, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.
.

Question 33 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11090
User-Defined ID:	RS20016-N01
Cross Reference Number:	BWLI-RY1-073
Topic:	RS20016-N01
Num Field 1:	2.9
Num Field 2:	2.9
Text Field:	SYS016 A3.01
Comments:	Braidwood NRC Exam 2009 Question # 33 016 Non-Nuclear Instrumentation System (NNIS) Ability to monitor automatic operation of the NNIS, including: Automatic selection of NNIS inputs to control systems TIER: 2 GROUP: 2 Cog Level: High Task No: R-RY-013 Obj No: 4C.RY-08 Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	Big Note RC-4 rev. 2 Bwd- Unit 1 PTLR rev. 4, figure 3.1
Justification for Non SRO CFR Link:	
Question History:	Previous version used in Bwd ILT systems phase.
Question Source:	Modified from Bwd ILT bank # BWLI-RY1-073
Low KA Justification (if required):	
Revision History:	modified pressure in stem from 550 to 650 psig. Changes answer from neither PORV to 1RY456 opening.
ILT	
Supplied Ref (If appropriate):	Bwd- Unit 1 PTLR rev. 4, figure 3.1
Excluded Reference:	

34 **ID: RS20041-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power.
- Steam pressure is 1005 psig.
- Steam Dump Mode Select switch is in the STEAM PRESSURE mode.
- 1PK-507, Steam Header Pressure Controller, is set at 6.93 (1039 psig).

The following then occurs:
- The U1 NSO initiates a load reduction to 50% power.

With the above conditions, U1 Steam Dumps will...

A. NOT open prior to reaching 50% power.

B. begin to OPEN at approximately 61% power.

C. begin to OPEN at approximately 68% power.

D. begin to OPEN at approximately 77% power.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to predict impact of changing steam pressure on steam dumps controls.
0% power corresponds to a Tave of 557° and steam pressure of 1092 psig.
100% power corresponds to a Tave of 587° and steam pressure of 1005 psig.
Using interpolation:
1039 psig corresponds to Tave of 575.3° and 61% power.

Choice A is incorrect, see explanation above.
Choice B is correct, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

Question 34 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11091
User-Defined ID:	RS20041-N01
Cross Reference Number:	BWLI-DU1-066
Topic:	RS20041-N01
Num Field 1:	3.1
Num Field 2:	3.2
Text Field:	SYS041 A1.02
Comments:	Braidwood NRC Exam 2009 Question # 34 041 Steam Dump System (SDS)/Turbine Bypass Control Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam pressure TIER: 2 GROUP: 2 Cog Level: High Task No: R-DU-001 Obj No: 3C.DU-01-A Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	ILT lesson plan I1-DU-XL-01 rev. 5a
Justification for Non SRO CFR Link:	
Question History:	Used in ILT systems.
Question Source:	ILT Bank # BWLI-DU1-066
Low KA Justification (if required):	
Revision History:	Modified from ILT Bank # BWLI-DU1-066
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

35 **ID: RS20045-N01** **Points: 1.00**

Given:
- Unit 1 is raising power from 5% to full load, following a 3 day forced outage.

In accordance with 1BwGP 100-3, POWER ASCENSION, Calorimetric Calculations must be performed at __(1)__%, __(2)__%, and __(3)__% power levels.

	(1)	(2)	(3)
A.	15	50	100
B.	30	75	90
C.	30	75	100
D.	50	90	100

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of administrative holds for performing NI calibrations during turbine load ascension.
Per 1BwGP 100-3 calorimetric surv. is performed and NIs adjusted at 50, 90 and 100% power.
15% distractor is minimum power level tech spec surv. requirement 3.3.1.2 (for calorimetric/NI adjustment) applies.
30% distractor was recently low power hold point for secondary chemistry.
75% distractor is maximum power level before surv. requirement 3.3.1.3 (for NI AFD comparison with incores) must be completed.
Choice A is incorrect, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is correct

Question 35 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11092
User-Defined ID:	RS20045-N01
Cross Reference Number:	BWLI-GP3-046
Topic:	RS20045-N01
Num Field 1:	2.6
Num Field 2:	2.9
Text Field:	SYS045 K4.27
Comments:	Braidwood NRC Exam 2009 Question # 35 045 Main Turbine Generator (MT/G) System Knowledge of MT/G system design feature(s) and/or interlock (s) which provide for the following: Calibrations of the nuclear instrumentation as flux shifts during T/G load increase (permissives and administrative holds) TIER: 2 GROUP: 2 Cog Level: Low Task No: R-NI-004 Obj No: 4C.NI-05 Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	1BwGP 100-3 rev. 47
Justification for Non SRO CFR Link:	
Question History:	used in ILT normal ops
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

36 **ID: RS20068-N01** **Points: 1.00**

Given:
- Unit 1 is in a refuel outage.
- 0A RHUT level has risen slowly for several hours due to RCS draining operations.
- Concurrently, the waste gas compressor was ran several times, maintaining waste gas vent header pressure between 1.5 psig and 2.0 psig.

With the above conditions, which of the following is an expected response?

A. ON COVER gas decay tank pressure has remained stable.

B. ON LINE gas decay tank pressure has lowered.

C. STANDBY gas decay tank pressure has lowered.

D. 0NT9318, N2 SUP HDR TO RHUT PCV has opened.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of cause effect relationship between liquid radwaste (RHUT) and waste gas header.
The RHUT are vented to the waste gas header.
When the RHUT level is raised the waste gas header pressure rises. As hdr pressure rises, the waste gas compressor is operated to maintain hdr pressure in normal operating range of 0.5 to 2.0 psig. Compressor discharges to on line GDT causing its pressure to rise.

Choice A is correct, the cover GDT will normally supply the gas make up to the waste gas header via an auto PCV, when header pressure drops below 1.3 psig. In this question, hdr pressure is maintained between 1.5 and 2.0 psig, so no use of the cover GDT is required and pressure should not change.
Choice B is incorrect, the online GDT receives discharge from compressor. Since compressor was ran, online GDT pressure will rise.
Choice C is incorrect, standby GDT auto aligns to the compressor discharge when the on line tank is full (95 psig). Therefore the standby tank pressure would only rise if the on line tank gets full, but pressure should never drop.
Choice D is incorrect, N2 make up to the RHUT (waste gas vent hdr) auto occurs at < 0.7 psig in the hdr. Pressure does not get that low.

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:	11093
User-Defined ID:	RS20068-N01
Cross Reference Number:	
Topic:	RS20068-N01
Num Field 1:	2.5
Num Field 2:	2.6
Text Field:	SYS068 K1.02
Comments:	Braidwood NRC Exam 2009 Question # 36 068 Liquid Radwaste System (LRS) Knowledge of the physical connections and/or cause effect relationships between the Liquid Radwaste System and the following systems: Waste gas vent header TIER: 2 GROUP: 2 Cog Level: High Task No: R-GW-008 Obj No: 3S.GW-01 Cross Ref: 10CFR55.41(b) (13)

General Data	
Technical Reference with Revision Number:	I1-GW-XL-01 rev. 3 Bwd ILT Gaseous Radwaste lesson plan.
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

37 **ID: RS20075-N01** **Points: 1.00**

Given:
- Both units are at 100% power.
- The following equipment is OOS.
  - 1B CW Pump
  - 0A WS Pump
  - U-0 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. ALL unit 1 steam dump valves close.

C. Unit 1 loses Instrument Air.

D. WS header pressure drops on BOTH units.

Answer: B

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 B 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 C is incorrect, U-0 SAC (which is OOS) is powered from bus 143, so the loss of bus 143 will not affect the status of instrument air.
Choice D 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.

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:	11094
User-Defined ID:	RS20075-N01
Cross Reference Number:	
Topic:	RS20075-N01
Num Field 1:	3.4
Num Field 2:	3.5
Text Field:	SYS075 K3.07
Comments:	Braidwood NRC Exam 2009 Question # 37 075 Circulating Water System Knowledge of the effect that a loss or malfunctions of the circulating water system will have on the following: ESFAS TIER: 2 GROUP: 2 Cog Level: High Task No: R-EF-001 Obj No: 3C.EF-01-B Cross Ref: 10CFR55.41(b)(7)

General Data	
Technical Reference with Revision Number:	Big note EF-1 rev.14 Big note AC-7 rev.6
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

38 **ID: RS20035-N01** **Points: 1.00**

Given:
- Unit 1 experienced a SG tube rupture at full power.
- Currently the crew is performing 1BwEP ES-3.1, POST SGTR COOLDOWN USING BACKFILL, at step 6, INITIATE RCS COOLDOWN TO 200°F.

The following note is prior to step 6.
"Since ruptured SG(s) may continue to depressurize to less than the minimum RCS pressure necessary for continued RCP operation, cooldown to cold shutdown should not be delayed."

The FIRST concern for RCP operation is...

A. excessive #1 seal leakoff.

B. net positive suction head (NPSH) requirements.

C. adequate #1 seal face gap.

D. causing an RCS pressure spike.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of operational implications of an EOP caution about the ruptured SG continuing to depressurize.
When cooling down the ruptured SG using backfill, the RCS pressure has to be lowered below the rupture SG pressure to achieve backflow and the desired result. If the ruptured SG pressure drops low enough that RCS has to be depressurized below approx. 215 psig (200 psid #1 seal DP) then the RCPs would have to be secured due to inadequate flow through the #1 seals leading to inadequate seal face gap.

Choice A is incorrect, the lowering RCS pressure would lead to lowering seal leak off, not excessive leak off.
Choice B is incorrect, NPSH requirements are typically not a concern for the RCPs. There are no cautions or statements in any of the operating procedures regarding RCP NPSH requirements.
Choice C is correct, see answer above.
Choice D is incorrect, RCS pressure spike is concern when starting an RCP with RCS temp <350°F and SG temp. > 50°F above RCS cold leg temps. (prohibited by notes in Tech Spec 3.4.6. and 3.4.7). However, with the RCP already running and the SGTR (RCS and SG are hydraulically coupled), pressure spike is not a concern.

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:	11095
User-Defined ID:	RS20035-N01
Cross Reference Number:	
Topic:	RS20035-N01
Num Field 1:	3.8
Num Field 2:	4.3
Text Field:	SYS035 2.4.20
Comments:	Braidwood NRC Exam 2009 Question # 38 035 Steam Generator System (S/GS) Knowledge of the operational implications of EOP warnings, cautions, and notes. TIER: 2 GROUP: 2 Cog Level: Low Task No: R-EP-004 Obj No: 3D.EP-16-B Cross Ref: 10CFR55.41(b)(10)

General Data	
Technical Reference with Revision Number:	1BwEP ES -3.1 rev.200 ILT lesson plan I1-EP-XL-04 rev. 17b ILT lesson plan I1-RC-XL-02 rev. 4b
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

39 **ID: RE10007-N01** **Points: 1.00**

Given:

- BOTH units were tripped from 100% power.
- 1/2BwEP ES-0.1, "REACTOR TRIP RESPONSE", are in progress.

Condition 1:
- BOTH units experience an RCS temperature drop to 540°F

Condition 2:
- BOTH units have two (2) control rods that did NOT fully insert.

Based on the above, which statement below is correct regarding the amount of boration required in each condition?

A. Unit 1 will require more boration for condition 1 AND Unit 2 will require more boration for condition 2.

B. Unit 2 will require more boration for condition 1 AND Unit 1 will require more boration for condition 2.

C. Unit 1 will require more boration for BOTH conditions.

D. Unit 2 will require more boration for BOTH conditions.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of procedural differences between units in 1/2BwEP ES-0.1.

Boration requirements for RCS temp < 545°F are:
Unit 1- 40 gal (120 gal from RWST) boric acid for each 1°F below 545°F
Unit 2- 35 gal (105 gal from RWST) boric acid for each 1°F below 545°F

Boration requirements for 2 or more rods not fully inserted are:
Unit 1- 1320 gal (5500 gal from RWST) boric acid for each rod not fully inserted.
Unit 2- 1200 gal (5000 gal from RWST) boric acid for each rod not fully inserted.

While examinees are not expected to memorize the exact numbers, both differences are a result of the Unit 1 RCS mass being larger than Unit 2, because of the S/G design differences. This one fact results in higher boration amounts always for unit 1.

Choice A is incorrect, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is correct, see explanation above.
Choice D is incorrect, see explanation above.

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:	11096
User-Defined ID:	RE10007-N01
Cross Reference Number:	
Topic:	RE10007-N01
Num Field 1:	3.8
Num Field 2:	3.9
Text Field:	EPE007 2.2.03
Comments:	Braidwood NRC Exam 2009 Question # 39 007 Reactor Trip (multi-unit license) Knowledge of the design, procedural, and operational | differences between units. TIER: 1 GROUP: 1 Cog Level: Low Task No: R-EP-035 Obj No: 3D.EP-01-C Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	1BwEP ES-0.1 rev. 200 2BwEP ES-0.1 rev. 200 ILT lesson plan I1-U1U2-XL-01 rev. 2a
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

40 **ID: RE10008-N01** **Points: 1.00**

Given:
- Unit 1 is operating at 100% power, normally aligned.
- 1RY8010C, PZR SAFETY VLV inadvertently lifts.

Thirty (30) seconds later:
- 1PI-469 PRT PRESS indicates 10 psig.
- 1PI-455A PZR PRESS indicates 1985 psig.

With the above indications, which of the following, most closely approximates the expected temperature on 1TI-464, SAF REL VLV C TEMP?

A. 150°F

B. 195°F

C. 245°F

D. 280°F

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of interrelations of Pzr vapor space accident and sensors (safety valve tailpipe temp)
Throttling steam thru safety valve is constant enthalpy process. Convert PRT pressure to psia (approx 25 psia). On Mollier diagram, move horizontally right from 2000 psia (Pzr press) until constant press line of 25 psia (interpolated) is met. Follow constant pressure line up to saturation curve, and interpolate where constant temperature curve meets saturation curve at the same point.
Choice A is incorrect, erred by following horizontal enthalpy line all the way to saturation curve.
Choice B is incorrect, erred by not converting PRT pressure to psia. (used 10 psia. curve)
Choice C is correct, see explanation above.
Choice D is incorrect, erred by following vertical entropy lines up to saturation curve (vs. following constant pressure curve.

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:	11097
User-Defined ID:	RE10008-N01
Cross Reference Number:	RE10008-004
Topic:	RE10008-N01
Num Field 1:	2.7
Num Field 2:	2.7
Text Field:	E/APE008 K2.02
Comments:	Braidwood NRC Exam 2009 Question # 40 008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Sensors and detectors TIER: 1 GROUP: 1 Cog Level: High Task No: R-OA-060 Obj No: 3D.OA-50-A Cross Ref: 10CFR55.41(b) (14)

General Data	
Technical Reference with Revision Number:	Steam Tables Mollier Diagram ILT lesson plan I1-TH-XL-03 rev. 8c
Justification for Non SRO CFR Link:	
Question History:	Previous version not used in Bwd ILT program
Question Source:	Modified from Bwd ILT Bank #RE10008-004
Low KA Justification (if required):	
Revision History:	Changed PRT pressure in stem from 20 to 10 psig. Added Prz press to stem.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

41 **ID: RE10009-N01** **Points: 1.00**

Given the following sequence of events:
- A medium break RCS LOCA occurred on loop 1A.
- 1BwEP ES-1.2 POST LOCA COOLDOWN AND DEPRESSURIZATION was in progress at step 1, when status tree monitoring identified a red path for Integrity due to 1A cold leg temperature and excessive cool down rate.
- The crew performed the actions of 1BwFR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK and 1A loop temperature has stabilized.

With the above conditions, which of the following describe the mitigation strategy that will be used for the LOCA?

A. Further RCS cooldown CANNOT commence until after a 1 hour soak. Subsequent cooldown must be <50°F in any 1 hour period.

B. Further RCS cooldown CAN commence immediately. The maximum allowable cooldown must be <50°F in any 1 hour period.

C. Further RCS cooldown CANNOT commence until after a 1 hour soak. Subsequent cooldown can be up to 200°F in any 1 hour period.

D. Further RCS cooldown CAN commence immediately ONLY in the unaffected loops. The 1A loop temperature must remain stable for 1 hour.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of actions to be taken if PTS limits are violated.
1BwFR-P.1 requires a 1 hour soak after RCS temperature has stabilized and then restricts subsequent cooldown to 50°F/hr.

Choice A is correct, see explanation above.
Choice B is incorrect, RCS must soak for 1 hour prior to cooldown commencing.
Choice C is incorrect, cooldown limit is 50°F/hr cooldown rate after PTS limits violated. 200°F/hr cooldown is the normal pressurizer cooldown limit.
Choice D is incorrect, 1BwFR-P.1 does not allow cooldown of unaffected loops. Mitigation strategy of other events such as SGTR perform loop specific actions, but it would not be appropriate for a PTS cooldown.

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:	11098
User-Defined ID:	RE10009-N01
Cross Reference Number:	
Topic:	RE10009-N01
Num Field 1:	3.8
Num Field 2:	4.4
Text Field:	E/APE009 A2.14
Comments:	Braidwood NRC Exam 2009 Question # 41 009 Small Break LOCA Ability to determine or interpret the following as they apply to a small break LOCA: Actions to be taken if PTS limits are violated TIER: 1 GROUP: 1 Cog Level: High Task No: R-FR-022 Obj No: 3D.FR-05-B Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	1BwFR-P.1 rev. 201
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

42 **ID: RE10015-N01** **Points: 1.00**

Given:
- A loss of CC flow to an RCP has occurred on Unit 1.
- 1BwOA RCP-2 LOSS OF SEAL COOLING was entered and the NSO is monitoring the RCP lower radial bearing temperature.

What RCP lower radial bearing temperature is listed in the Operator Action Summary of 1BwOA RCP-2 as requiring a reactor trip?

A. >150°F

B. >184°F

C. >225°F

D. >235°F

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of when to secure RCPs on high bearing temperature.
1BwOA RCP-2 Operator Action Summary requires RCP trip when lower radial bearing temp rises >225°F.

Choice A is incorrect, 150°F is temperature at which CC flow must be maintained to thermal barrier in the event of a loss of seal injection to prevent seal damage.
Choice B is incorrect, 184°F is the lower radial bearing high temp alarm setpoint.
Choice C is correct, see explanation above.
Choice D is incorrect, 235°F is the seal outlet temp trip setpoint.

Question 42 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:	11099
User-Defined ID:	RE10015-N01
Cross Reference Number:	BWLI-RC2-031
Topic:	RE10015-N01
Num Field 1:	3.4
Num Field 2:	3.5
Text Field:	E/APE015 A2.08
Comments:	Braidwood NRC Exam 2009 Question # 42 015/017 Reactor Coolant Pump (RCP) Malfunctions Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on high bearing temperature TIER: 1 GROUP: 1 Cog Level: Low Task No: R-OA-061 Obj No: 4D.OA-51 Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	1BwOA RCP-2 rev. 100
Justification for Non SRO CFR Link:	
Question History:	Not currently used in ILT.
Question Source:	Bwd ILT Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

43 **ID: RE10022-N01** **Points: 1.00**

Given:
- Unit 2 is at 75% power, MOL, normally aligned.
- 2CV182, CHG HDR BACK PRESS CONTROL VLV controller has failed to zero demand.
- The crew is performing 2BwOA PRI-15 LOSS OF NORMAL CHARGING.
- Letdown has been isolated.
- A VCT automatic makeup is occurring.

When the NSO verifies proper RMCS operation, which of the following valve alignments is correct?

	BORIC ACID TO BLNDR 2CV110A	PW TO BORIC ACID BLNDR 2CV111A	BORIC ACID BLNDR TO CHG PMPS 2CV110B	BORIC ACID BLNDR TO VCT 2CV111B
A.	throttled	closed	open	closed
B.	throttled	throttled	open	closed
C.	closed	throttled	closed	open
D.	closed	throttled	open	open

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to determine MCR indications correctly reflect desired lineup during loss of Rx Coolant Makeup.

Choice A is incorrect, this is line up for RMCS in Borate mode.
Choice B is correct alignment for Automatic mode.
Choice C is incorrect, this is line up for RMCS in Dilute mode.
Choice D is incorrect, this is line up for RMCS in Alt. Dilute mode.

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:	11100
User-Defined ID:	RE10022-N01
Cross Reference Number:	
Topic:	RE10022-N01
Num Field 1:	4.6
Num Field 2:	4.3
Text Field:	E/APE022 2.1.31
Comments:	Braidwood NRC Exam 2009 Question # 43 022 Loss of Reactor Coolant Makeup Ability to locate control room switches, controls, and indications, and to | determine that they correctly reflect the desired plant lineup. TIER: 1 GROUP: 1 Cog Level: Low Task No: R-CV-006 Obj No: 3C.CV-06-B Cross Ref: 10CFR55.41(b) (6)

General Data	
Technical Reference with Revision Number:	Big note CV-3 rev.12
Justification for Non SRO CFR Link:	
Question History:	used in ILT system phase
Question Source:	ILT Bank
Low KA Justification (if required):	
Revision History:	revised question form essay to multiple choice
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

44 **ID: RE10025-N01** **Points: 1.00**

Given:
- Unit 2 is in Mode 5.
- RCS pressure is 100 psig.
- 2A CV pump is in operation.
- VCT pressure is 15 psig.
- U-0 CC Pump and HX are mechanically aligned to Unit 2.
- 2A RH pump is running in the shutdown cooling mode.
- Unit 2 FC cooling loop is in operation.

The following MCR indications are noted:
- Annunciator 2-2-A5, CC SURGE TANK LEVEL HIGH LOW alarms.
- CC Surge Tank level indication is 65% and slowly RISING.

With the above conditions, a tube leak in which one of the following heat exchangers would cause the CC surge tank level rise?

A. 2A RH Hx

B. U-2 FC Hx

C. U-0 CC Hx

D. U-2 Seal Water Hx

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to determine leakage of RHR into closed loop cooling system (into CC system).
2BwOA PRI-6, attachment B, lists sources of CC inleakage and outleakage. With CC surge tank level rising the only choice which would cause CC inleakage is the online RH HX. CC system pressure is typically 130-140 psig at the CC pump discharge and slightly lower at the system loads (depending on flow and system head loses). RH system pressure at the RH Hx is approx. 260 psig. The RH pump head will add about 160# to suction pressure (RCS pressure) at normal flow. Therefore rising CC surge tank level would indicate an RH Hx leak.

Choice A is correct, see explanation above.
Choice B is incorrect, FC Hx is listed as potential source of CC outleakage per 2BwOA PRI-6. FC pressure is normally 50-90 psig (lower than CC pressure)
Choice C is incorrect, U-0 CC Hx is listed as potential source of CC outleakage per 2BwOA PRI-6. SX pressure is normally 80-110 psig (lower than CC pressure)
Choice D is incorrect, Seal water Hx is listed as potential source of CC outleakage per 2BwOA PRI-6. Seal return pressure is determined by backpressure of VCT. At 15 psig VCT pressure, seal return pressure will be lower than CC pressure.

Question 44 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:	11104
User-Defined ID:	RE10025-N01
Cross Reference Number:	
Topic:	RE10025-N01
Num Field 1:	3.4
Num Field 2:	3.8
Text Field:	E/APE025 A2.02
Comments:	Braidwood NRC Exam 2009 Question # 44 025 Loss of Residual Heat Removal System (RHRS) Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Leakage of reactor coolant from RHR into closed cooling water system or into reactor building atmosphere TIER: 1 GROUP: 1 Cog Level: High Task No: R-RH-004 Obj No: 4C.RH-05 Cross Ref: 10CFR55.41(b)(7)

General Data	
Technical Reference with Revision Number:	2BwOA PRI-6 rev. 102 Bwd Big note CC-2 rev. 7
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New Note: Several Bwd ILT bank questions are similar to this question. However, all bank questions ask for source of CC outleakage with normal at power configuration.
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

45 **ID: RE10026-N01** **Points: 1.00**

Given:
- Unit 1 is in a refuel outage
.
- Unit 2 is in Mode 4, in a forced outage.
- 2B RH train is in shutdown cooling.

- The Unit 0 CC HX is aligned to Unit 1 supplying BOTH 1A and 2A RH loops.

The following occurs:
- A large leak in the Unit 2 CC system causes both U-2 CC pumps to trip.
- The crew enters 2BwOA PRI-6, COMPONENT COOLING MALFUNCTION.

In the current system configuration, which of the following describes the preferred mitigation strategy for RCS heat removal?

A. Continue 2B RH train operation.

B. Secure 2B RH shutdown cooling AND steam the SGs.

C. Secure 2B RH shutdown cooling AND initiate bleed and feed.

D. Swap shutdown cooling to the 2A RH train.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of shutdown implications of loss of CC mitigation strategies.
With the unit 0 CC Hx aligned to unit 1, CC would still be available to the 2A RH loop. Therefore, RH cooling could be swapped to the 2A RH train.

Choice A is incorrect, would be correct if the 2B RH loop was supplied from U-1 (U-0 HX), but only 2A loop can be supplied from U-1.
Choice B is incorrect, would be correct if no RH Hx had cooling (2A still has cooling).
Choice C is incorrect, would be correct if no RH Hx had cooling and steaming the SGs was not an option.
Choice D is correct, 2BwOA PRI-10 will start the redundant RH train (if available) prior to establishing alternate core cooling.

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:	11101
User-Defined ID:	RE10026-N01
Cross Reference Number:	
Topic:	RE10026-N01
Num Field 1:	3.8
Num Field 2:	4.2
Text Field:	E/APE026 2.4.09
Comments:	Braidwood NRC Exam 2009 Question # 45 026 Loss of Component Cooling Water (CCW) Knowledge of low power/shutdown implications in accident (e.g., loss of coolant | accident or loss of residual heat removal) mitigation strategies. TIER: 1 GROUP: 1 Cog Level: High Task No: R-OA-027 Obj No: 3D.OA-10-A Cross Ref: 10CFR55.41(b) (4)

General Data	
Technical Reference with Revision Number:	Big note CC-1 rev. 11 2BwOA PRI-6 rev.102 2BwOA PRI-10 rev. 102
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

46 **ID: RE10027-NO1** **Points: 1.00**

Given:
- Unit 1 is at 100% power.
- Pzr pressure control is automatic.
- PRESS CONT CH SELECT switch is in 457/458 position at 1PM05J.
- Pzr pressure transmitter 1PT-457 has failed full scale HIGH.
- No operator action is taken.

Which ONE of the following describes the initial response of the Pzr PORVs?

A. 1RY-455A remains closed and 1RY-456 opens.

B. 1RY-455A opens and 1RY-456 remains closed.

C. BOTH PORVs remain closed.

D. BOTH PORVs open.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to monitor proper operation of Prz PORVs during pressure control channel failure.
Failing full scale high is equivalent to 2500#. With Prz press control in the 457/458 position, 1PT-457 failure would open 1RY-455A until pressure dropped to 2185#. At 2185# the interlock from 1PT-458 would then close the PORV.
1PT-457 is the interlock to 1RY-456, however, 1PT-457 failing high would have no affect to open 1RY-456.

Choice A is incorrect, see explanation above
Choice B is correct, see explanation above
Choice C is incorrect, see explanation above
Choice D is incorrect, see explanation above

Question 46 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11102
User-Defined ID:	RE10027-NO1
Cross Reference Number:	BWLI-INS2050
Topic:	RE10027-NO1
Num Field 1:	4.0
Num Field 2:	3.9
Text Field:	E/APE027 A1.01
Comments:	Braidwood NRC Exam 2009 Question # 46 027 Pressurizer Pressure Control System (PZR PCS) Malfunction Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: PZR heaters, sprays, and PORVs TIER: 1 GROUP: 1 Cog Level: High Task No: R-OA-100 Obj No: 3D.OA-11-A Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	Bwd Big note RY-2 rev. 9
Justification for Non SRO CFR Link:	
Question History:	Not currently used in ILT
Question Source:	ILT Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

47 **ID: RE10029-N01** **Points: 1.00**

Given:
- Unit 1 has experienced an ATWS condition and the crew is performing 1BwFR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, at step 2, VERIFY TURBINE TRIP.
- Attempts to manually trip the turbine have been UNSUCCESSFUL from the trip pushbutton and the OWS graphic 5512.

With the above conditions, per 1BwFR-S.1, the NSO will run back the turbine...

A. at OWS graphic 5501 by selecting MANUAL, then selecting RAPID, then selecting and holding GV lower arrow.

B. at 1PM02J by depressing 1HS-EH019 CD/FW runback pushbutton.

C. at OWS graphic 5501 by programming a REF DEMAND of 0, then programming a LOAD RATE of 250 MW/MIN, then selecting GO.

D. at OWS graphic 5512 by selecting START CD/FW RUNBACK.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to operate manual trip of main turbine during an ATWS condition.
The choices are 4 different methods to runback the turbine, however, 1BwFR-S.1 immediate actions step 2, RNO column, has the operator runback the turbine in the fastest manual method that will completely close the GVs. With the normal manual trip methods unsuccessful the action to run back the turbine until GVs are full closed will induce an anti-motoring turbine trip. The term "run back" is used in the question stem because "run back" is also used in 1BwFR-S.1, step 2 RNO, and it lends credibility to the distractors.

Choice A is correct per 1BwFR-S.1
Choice B is incorrect, this runback method is used in 1BwOA SEC-1, step 4 action column, but will not completely close the all GVs
Choice C is incorrect, this runback method is used in 1BwOA SEC-1, step 4 RNO column, but with a different REF DEMAND.
Choice D is incorrect, this runback method is used in 1BwOA SEC-1, step 4 action column, but will not completely close the all GVs

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:	11105
User-Defined ID:	RE10029-N01
Cross Reference Number:	
Topic:	RE10029-N01
Num Field 1:	4.1
Num Field 2:	3.9
Text Field:	E/APE029 A1.13
Comments:	Braidwood NRC Exam 2009 Question # 47 029 Anticipated Transient Without Scram (ATWS) Ability to operate and monitor the following as they apply to a ATWS: Manual trip of main turbine TIER: 1 GROUP: 1 Cog Level: Low Task No: R-FR-018 Obj No: 3D.FR-01-B Cross Ref: 10CFR55.41(b)(10)

General Data	
Technical Reference with Revision Number:	1BwFR-S.1 rev. 201
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

48 **ID: RE10055-N01** **Points: 1.00**

Given:
- BOTH units are at 100% power normal alignment.

The following sequence of events occurs:
- Lightning in the switchyard faults Unit 1 SATs.
- Unit 1 trips and loses ALL AC power.
- Both Unit 1 ESF buses are cross tied to Unit 2.
- The NSO is currently crosstying bus 144 to 142 per 1BwCA-0.1, LOSS OF ALL AC POWER RECOVERY WITHOUT SI REQUIRED.

The MAJOR reason for cross tying bus 144 and bus 142 is to energize the...

A. 0A fire pump.

B. 0B WS pump.

C. Unit 1 SAC.

D. pressurizer heaters.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of reasons for actions contained in EOP for station blackout (1BwCA-0.1).

Choice A is incorrect, in normal alignment the 0B fire pump (diesel) is still available for station needs.
Choice B is incorrect, 0C WS pump is energized from U-2.
Choice C is incorrect, Unit 2 SAC is available for station needs.
Choice D is correct, pressurizer heaters are needed to control unit 1 Prz pressure and avoid a unit 1 SI.

Question 48 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:	11103
User-Defined ID:	RE10055-N01
Cross Reference Number:	
Topic:	RE10055-N01
Num Field 1:	4.3
Num Field 2:	4.6
Text Field:	E/APE055 K3.02
Comments:	Braidwood NRC Exam 2009 Question #48 055 Loss of Offsite and Onsite Power (Station Blackout) Knowledge of the reasons for the following responses as the apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power TIER: 1 GROUP: 1 Cog Level: Low Task No: R-CA-001 Obj No: 3D.CA-09-B Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	1BwCA-0.1 rev. 201, step 12
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

49 **ID: RE10056-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power, normal alignment.

- A loss of off site power occurs.

Time	Event
0 sec.	1A DG re-energizes bus 141.
+17 sec.	(now) The NSO is monitoring ESF equipment sequencing.

From the following, which load will be sequenced on to bus 141 NEXT, and why?

A. 1A CV pump to ensure RCS inventory control.

B. 1A CC pump to ensure cooling of potentially contaminated equipment.

C. 1A AF pump to ensure a secondary heat sink.

D. 1A SX pump to ensure cooling of ESF equipment.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of reason for order and timing of LOOP sequencer.
The sequencing order from the time the bus is re-energized is:
0 sec-CV pps & VC WO pps.
5 sec-SI pumps (only if accompanied by SI)
10 sec-RH pumps (only if accompanied by SI)
15 sec-VC chiller
15-18 sec-CS pump (if CS signal is present)
20 sec- CC pumps
25 sec-SX pumps
35 sec-AF pumps

Question stem is sequencing scenario without a SI. Current time is 17 sec. after sequencer starts. Next start should be CC pp.
Choice A is incorrect, see explanation above.
Choice B is correct, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

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:	11106
User-Defined ID:	RE10056-N01
Cross Reference Number:	
Topic:	RE10056-N01
Num Field 1:	3.5
Num Field 2:	3.9
Text Field:	E/APE056 K3.01
Comments:	Braidwood NRC Exam 2009 Question # 49 056 Loss of Offsite Power Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power: Order and time to initiation of power for the load sequencer TIER: 1 GROUP: 1 Cog Level: low Task No: R-DG-003 Obj No: 3C.DG-01-B Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	ILT lesson plan I1-DG-XL-01 rev. 5a
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

50 **ID: RE10058-N01** **Points: 1.00**

Given:
- 1A DG is running in EMERGENCY mode and carrying the ESF bus loads following an undervoltage on bus 141.
- Subsequently, a loss of DC bus 111 occurs.

In response to the loss of DC, the 1A DG will...

A. NOT trip and continue to supply bus 141 loads.

B. trip due to an overspeed condition.

C. trip due to combustion air butterfly valve closing.

D. trip due to fuel control cylinder expanding to "OFF" position.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of reason for response of DG during loss of DC control power.
In emergency mode, all DG trips except three are blocked by the emergency fuel control solenoids (EFCS) being closed. However, a loss of DC control power fails the EFCSs open and the pneumatic fuel control valve open, which results in pneumatic protection air repositioning the fuel control cylinder to "OFF".

Choice A is incorrect, see explanation above.
Choice B is incorrect, loss of DC control power will fail fuel racks closed making overspeed impossible.
Choice C is incorrect, the combustion air butterfly valve is normally latched open. It can be mechanically closed by a link to the overspeed governor, but no reason for overspeed governor to actuate on a loss of DC.
Choice D is correct, see explanation above.

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:	11107
User-Defined ID:	RE10058-N01
Cross Reference Number:	
Topic:	RE10058-N01
Num Field 1:	3.4
Num Field 2:	3.7
Text Field:	E/APE058 K3.01
Comments:	Braidwood NRC Exam 2009 Question # 50 058 Loss of DC Power Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of dc control power by D/Gs TIER: 1 GROUP: 1 Cog Level: low Task No: R-DG-003 Obj No: 3C.DG-01-B Cross Ref: 10CFR55.41(b)(7)

General Data	
Technical Reference with Revision Number:	Bwd big note DG-6 rev. 8 ILT lesson plan I1-DG-XL-01 rev. 5a
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

51 **ID: RE10065-N01** **Points: 1.00**

Given:
- Unit 2 is at 75% power, normal alignment.
- The 2A Regen Heat Exchanger is in service.

The following sequence of events occurs:
- A control power fuse for 2IA066, INSTRUMENT AIR INSIDE CNMT ISOL VLV, blows in 2PM11J.
- Instrument Air to Unit 2 containment is isolated and ALL CVCS valves are in their fail position.
- Five (5) minutes later, the defective fuse is replaced, and 2IA066 is re-opened.
- Instrument air is fully restored to Unit 2 containment.

NO other operator actions were taken. (i.e. assume all CVCS valve control switches have NOT been moved from their normal at power alignment)

With the above conditions, the NSO monitoring 2PM05J will note which of the following letdown valves has automatically re-opened.

A. 2CV459, LTDWN LINE ISOL VLV

B. 2CV8389A, LTDWN TO REGEN HX 2A ISOL VLV

C. 2CV8149A, 45 GPM LTDWN ORIF 2A ISOL VLV

D. 2CV8160, LTDWN LINE CNMT ISOL VLV

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to monitor letdown system as IA is restored to the system.
All of the choices fail closed on loss of inst. air to cnmt. However, only 2CV8389A control switch is normally in the "open" position. The three distractors all have control switches that spring return to "auto" (because they have auto close functions of either phase A or low pressurizer level). 2CV8389A interlock to open is that the corresponding charging isol valve to regen hx (2CV8324A) must be open. 2CV8324A has no interlocks and it's C/S also maintains in the open position. Therefore, when air is restored, 2CV8324A opens first, then 2CV8289A opens. The three distractors must all have their C/S re-positioned to "open" before the valves will re-open.

Choice A is incorrect, see explanation above.
Choice B is correct, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

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:	11108
User-Defined ID:	RE10065-N01
Cross Reference Number:	
Topic:	RE10065-N01
Num Field 1:	2.9
Num Field 2:	3.1
Text Field:	E/APE065 A1.03
Comments:	Braidwood NRC Exam 2009 Question # 51 065 Loss of Instrument Air Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Restoration of systems served by instrument air when pressure is regained TIER: 1 GROUP: 1 Cog Level: High Task No: R-IA-002 Obj No: 3C.IA-01-A Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	Big note CV-1 rev. 11 20-E-2-4030CV20, 24, 27, 28 & 29
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

52 **ID: RE10077-N01** **Points: 1.00**

Given:
- Braidwood Station has been notified that the State Estimator alarm predicts a potential degraded grid condition in the event of a Braidwood unit trip.
- The crew is performing 0BwOA ELEC-1, ABNORMAL GRID CONDITIONS, ATTACHMENT A, STEP 1, CALCULATE UNIT 1 4KV NON-ESF BUS LOADING.

Which of the following parameters are MULTIPLIED TOGETHER to calculate NON-ESF BUS LOADING?

A. bus volts and feed breaker amps

B. UAT watts and bus volts

C. feed breaker amps and generator VARS

D. generator VARS and UAT watts

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of terms definitions (volts times amps equals load/Kva).
0BwOA ELEC-1 calculates bus loading using bus volts and feed brkr amps. These meters are read from 1PM01J. The distractors (gen VARS and UAT watts) are also on 1PM01J meters but are not used in the calculation. While it would not be expected that the examinee know which meters are used in the procedure from memory, he/she should be able to deduce the correct answer given their previous knowledge of electrical theory and term definitions.

Choice A is correct, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

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:	11109
User-Defined ID:	RE10077-N01
Cross Reference Number:	
Topic:	RE10077-N01
Num Field 1:	3.3
Num Field 2:	3.5
Text Field:	E/APE077 K1.01
Comments:	Braidwood NRC Exam 2009 Question # 52 077 Generator Voltage and Electric Grid Disturbances Knowledge of the operational implications of the following concepts as | they apply to Generator Voltage and Electric Grid Disturbances: Definition of terms: volts, watts, amps, VARs, power factor TIER: 1 GROUP: 1 Cog Level: low Task No: R-OA-128 Obj No: 4D.OA-91 Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	0BwOA ELEC-1 rev. 7
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

53 **ID: RE1WE04-N01** **Points: 1.00**

Given the following sequence of events:
- Unit 1 was at 100% power with ESF bus 142 de-energized due to a bus fault.
- Subsequently, a manual reactor trip and SI occur due to a pressurizer safety valve stuck partially open.
- The MCR receives a report that a large amount of water is leaking in the U-1 containment penetration area.
- The crew transitions to 1BwCA-1.2, LOCA OUTSIDE CONTAINMENT, with all 1A train ECCS pumps still operating.

The following indications are CURRENTLY noted on the MCB:
- RCS pressure is 1700 psig and stable.
- 1A RH discharge flow is 0 gpm.
- 1A SI pump discharge flow is 200 gpm.
- 1A CV pump flow is 300 gpm.

The leak can be reduced by closing...
(assume the leak is ALL RWST water)

A. 1SI8801A, CHG PMPS TO COLD LEG INJ ISOL VLV and stopping the 1A CV pump.

B. 1SI8835, SI PMPS TO COLD LEG ISOL VLV and stopping the 1A SI pump.

C. 1SI8809A, RH TO COLD LEG 1A & 1D ISOL VLV and stopping the 1A RH pump.

D. 1SI8809B, RH TO COLD LEG 1B & 1C ISOL VLV and stopping the 1A RH pump.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of interrelations of a LOCA outside containment and proper operation of emergency heat removal system (ECCS). Shut off head for the ECCS pumps are as follows: RH-200 psid, SI-1500 psid, CV-2600 psid. At the current RCS pressure of 1700 psid the only pump that should have indicated flow is the CV pump. Therefore, if the SI pump indicates 200 gpm and assuming the leak is RWST water, it is logical to assume that the leak is somewhere on the SI pump discharge line. 1BwCA-1.2 step 2.d, isolates each of the ECCS pump discharge paths one at a time and checks for indication the leak has stopped.

Choice A is incorrect, CV pump flow is normal for current RCS pressure.
Choice B is correct, see explanation above.
Choice C is incorrect, RH pump flow is normal for current RCS pressure.
Choice D is incorrect, RH pump flow is normal for current RCS pressure.

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:	11110
User-Defined ID:	RE1WE04-N01
Cross Reference Number:	
Topic:	RE1WE04-N01
Num Field 1:	3.8
Num Field 2:	4.0
Text Field:	W/E04 K2.02
Comments:	Braidwood NRC Exam 2009 Question # 53 E04 LOCA Outside Containment Knowledge of the interrelations between the (LOCA Outside Containment) and the following: Facility*s heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. TIER: 1 GROUP: 1 Cog Level: high Task No: R-CA-004 Obj No: 4D.CA-03 Cross Ref: 10CFR55.41(b) (8)

General Data	
Technical Reference with Revision Number:	Big note ECCS-1 rev.9
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

54 **ID: RE1WE05-N01** **Points: 1.00**

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 1BwFR-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. Containment pressure rises to 5.2 psig.

C. ALL SG WR levels drop to 20%.

D. 1A CV pumps trips.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of conditions signals (MCR indications) that would require bleed and feed (operational implication)
1BwFR-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 A is incorrect, pressure still below 2335#.
Choice B is incorrect, containment would be adverse, but current SG levels at 50% would not meet bleed and feed criteria.
Choice C is correct, levels are below normal values of 27%.
Choice D is incorrect, 1B CV pp would still be running.

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:	11111
User-Defined ID:	RE1WE05-N01
Cross Reference Number:	
Topic:	RE1WE05-N01
Num Field 1:	3.9
Num Field 2:	4.1
Text Field:	W/E05 K1.03
Comments:	Braidwood NRC Exam 2009 Question # 54 E05 Loss of Secondary Heat Sink Knowledge of the operational implications of the following concepts as they apply to the (Loss of Secondary Heat Sink): Annunciators and conditions indicating signals, and remedial actions associated with the (Loss of Secondary Heat Sink). TIER: 1 GROUP: 1 Cog Level: High Task No: R-FR-029 Obj No: 4D.FR-08 Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	1BwFR-H.1 rev. 200
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

55 **ID: RE1WE11-N01** **Points: 1.00**

Given :
- An RCS LOCA occurred on Unit 1.
- 1SI8811A/B, CNMT SUMP ISOL VLVs, are CLOSED and CANNOT be opened.
- 1BwCA-1.1 "LOSS OF EMERGENCY COOLANT RECIRCULATION" was entered.
- Minimum ECCS flow to remove decay heat has been established per 1BwCA-1.1.

With the above conditions and in accordance with 1BwCA-1.1, the NSO monitoring plant parameters will raise RCS make-up flow if...

A. pressurizer level drops to 0%.

B. RVLIS HEAD region drops to 31%.

C. hot leg wide range temperatures rise by 5°F.

D. core exit thermocouples rise by 5°F.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of operational implications (when to raise RCS make-up) of specific components associated with loss of emergency coolant recirc.
1BwCA-1.1 Continuous Action Summary requires the operator to raise RCS make-up flow if either the CETCs rise or if the RVLIS plenum region drops below 15%. This is less restrictive than criteria used in other emergency procedures to raise ECCS flow because of the objective to conserve RWST water.

Choice A is incorrect, pressurizer level is used in determining SI reinitiation criteria in other emergency procedures (1BwEP-3), but not in 1BwCA-1.1
Choice B is incorrect, RVLIS plenum region criteria for raising flow is 15%.
Choice C is incorrect, hot leg wide range temperatures are not used because they can be unreliable of core conditions during a LOCA due to specific RCS flow conditions.
Choice D is correct, see explanation above.

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:	11112
User-Defined ID:	RE1WE11-N01
Cross Reference Number:	
Topic:	RE1WE11-N01
Num Field 1:	3.7
Num Field 2:	4.0
Text Field:	W/E11 K1.01
Comments:	Braidwood NRC Exam 2009 Question # 55 E11 Loss of Emergency Coolant Recirculation Knowledge of the operational implications of the following concepts as they apply to the (Loss of Emergency Coolant Recirculation): Components, capacity, and function of emergency systems. TIER: 1 GROUP: 1 Cog Level: Low Task No: R-CA-003 Obj No: 3D.CA-02-B Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	1BwCA-1.1 rev.202
Justification for Non SRO CFR Link:	
Question History:	none
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

56 **ID: RE1WE12-N01** **Points: 1.00**

Given:
- 10 minutes ago Unit 2 experienced a reactor trip and SI due to an unisolable fault of ALL steam generators.
- Bus 242 is faulted.
- The 2A AF pump has tripped on overcurrent.
- 2BwCA-2.1"UNCONTROLLED DEPRESSURIZATION OF ALL STEAM GENERATORS" is in progress.
- RCS Cold leg temperatures are 425°F.
- ALL SG narrow range levels indicate 0%.
- The MCR valve controller for 2AF005H, 2D SG FLOW CONT VLV, is failed in the full demand (open) position.
- All OTHER equipment operates as designed.

With the above conditions, to meet the 2D SG AF flow requirement of 2BwCA-2.1, the...

A. 2AF005H valve can be throttled at the RSP or locally.

B. 2AF005H valve can ONLY be throttled at the RSP.

C. 2AF013D, SG 2D ISOL VLV, can be throttled from the MCR.

D. 2AF013H, SG 2D ISOL VLV, can be throttled from the MCR.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of the interrelationship between a uncontrolled depressurization of all SGs and the manual features of the AF system.
The 2AF005 valves can be controlled with IA controllers from the RSP that bypass the electrical signal from the MCR controllers. Therefore the valve can be controlled from the RSP. Also the 2AF005 valves have local handwheels that allow local throttling or closing. With the given information 2BwCA-2.1 would require 45 gpm flow be established to each SG. With 2A AF pp not available and bus 142 faulted (B train 2AF013 valves without power) the 2AF005H valve would need to be throttled either at the RSP or locally.

Choice A is correct, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, 2AF013D is A train and 2A AF pump is tripped.
Choice D is incorrect, 2AF013H does not have power with bus 242 de-energized.

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:	11113
User-Defined ID:	RE1WE12-N01
Cross Reference Number:	
Topic:	RE1WE12-N01
Num Field 1:	3.4
Num Field 2:	3.7
Text Field:	W/E12 K2.01
Comments:	Braidwood NRC Exam 2009 Question # 56 E12 Uncontrolled Depressurization of all Steam Generators Knowledge of the interrelations between the (Uncontrolled Depressurization of all Steam Generators) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. TIER: 1 GROUP: 1 Cog Level: High Task No: R-CA-005 Obj No: 3D.CA-04-B Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	I1-AF-XL-01 rev. 4b Bwd ILT Aux. Feed lesson plan.
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

57 **ID: RE10003-N01** **Points: 1.00**

Given:
- Unit 1 was at 95% power normal alignment, ramping to 100%, when the following occurs:
  - During an automatic rod step, Control Bank D, Rod M-12 drops part way into the core.
  - Rod M-12 DRPI now indicates 120 steps.
  - All other CB-D DRPI indications are 198 steps.
  - DRPI ACCURACY MODE SWITCH is in A+B position.
  - The crew enters 1BwOA ROD-3, DROPPED OR MISALIGNED ROD, and determines that the failure occurred in rod control, but NOT DRPI.

Based on the above conditions, the LOWEST possible ACTUAL position of Rod M-12 is _____ steps.

A. 108

B. 110

C. 116

D. 120

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee ability to determine actual rod position vs. indication during a dropped control rod.
DRPI position accuracy is normally ± 4 steps (3 steps + 1 step for manufacturing tolerances) when in the A+B mode.

Choice A is incorrect, but plausible because Tech Spec LCO requires ± 12 steps.
Choice B is incorrect, would be correct in half accuracy mode A-only.
Choice C is correct, see explanation above.
Choice D is incorrect, is indicated rod position without accounting for tolerances.

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:	11114
User-Defined ID:	RE10003-N01
Cross Reference Number:	
Topic:	RE10003-N01
Num Field 1:	3.7
Num Field 2:	3.9
Text Field:	E/APE003 A2.01
Comments:	Braidwood NRC Exam 2009 Question # 57 003 Dropped Control Rod Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod position indication to actual rod position TIER: 1 GROUP: 2 Cog Level: High Task No: R-PI-001 Obj No: 3C.PI-01-A Cross Ref: 10CFR55.41(b)(6)

General Data	
Technical Reference with Revision Number:	Big note RD-6 rev. 7
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

58 **ID: RE10037-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power, normal alignment.
- A tube leak develops on the 1C S/G and the crew enters 1BwOA SEC-8, STEAM GENERATOR TUBE LEAK.
- Leak rate conditions have stabilized, and the NSO is determining the tube leak rate.
- The following are indications on 1PM05J.

- 1FI-121A, CHG HDR FLOW	94.0 gpm
- 1FI-132, LTDWN LINE FLOW	75.0 gpm
- 1FI-142A thru 145A, RCP 1_ SEAL WTR FLOW	8.5 gpm per RCP
- 1FT-158 thru 161, RCP 1_ SEAL LEAKOFF FLOW	3.4 gpm per RCP
- 1LI-459A PZR LEVEL	60% and stable
- Excess Letdown	Isolated.

Based on the above indications, U1 Primary to Secondary leakage is ____ gpm.

A. 5.4

B. 19.0

C. 39.4

D. 53.0

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to determine flow rate of leak. Flow rate is determined by observing difference between charging flow and seal leakoff plus letdown flow.

Choice A is correct, 94 - [75 + 3.4(4)] = 5.4
Choice B is incorrect, 94 - 75 = 19.0 (incorrectly neglects seal return flow)
Choice C is incorrect, [94 + 8.5(4)] - [75 + 3.4(4)] = 39.4 (incorrectly adds seal injection flow to charging.
Choice D is incorrect, [94 + 8.5(4)] - 75 = 53.0 (incorrectly adds seal injection flow to charging AND incorrectly neglects seal return flow).

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:	11115
User-Defined ID:	RE10037-N01
Cross Reference Number:	
Topic:	RE10037-N01
Num Field 1:	4.3
Num Field 2:	4.4
Text Field:	E/APE037 2.1.23
Comments:	Braidwood NRC Exam 2009 Question # 58 037 Steam Generator (S/G) Tube Leak Ability to perform specific system and integrated plant procedures during all | modes of plant operation. TIER: 1 GROUP: 2 Cog Level: High Task No: R-OA-003 Obj No: 3D.OA-17-B Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	1BwOA SEC-8 rev.103
Justification for Non SRO CFR Link:	
Question History:	Used in Bwd ILT NOPS phase
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	Revise flow indications and distractors to make distractors match common calculation errors.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

59 **ID: RE10059-N01** **Points: 1.00**

Given:
- Liquid radwaste is being processed through the radwaste portable demineralizers in the truck bay.
- A significant leak develops at the demineralizer inlet, spraying radioactive water into the surrounding area.
- The leak has caused an ALERT condition alarm at the MCR RM-11.

Which ONE of the following radiation monitors will indicate this ALERT condition?

A. 0PR11J, RADWASTE EVAP CUBICLE

B. 0PR26J, RADWASTE VENT

C. 1PR28J, UNIT 1 AUX BLDG VENT STACK EFFLUENT

D. 0AR55J, FHB FH INCIDENT

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of the interrelations between a radwaste liquid release (leak) and the radioactive gas monitors.
The radwaste portable demins are located in the solid radwaste bldg. truck bay. The radwaste processing system has been significantly modified since the original plant construction, and location of components is diverse through the RCA. This question is difficult to answer based on casual knowledge or assumptions.

Choice A is incorrect, the radwaste evaporators were originally designed to handle radwaste processing, but are now bypassed when processing through the vendor demins.
Choice B is correct, the 0PR26J monitors exhaust from the solid radwaste bldg.
Choice C is incorrect, the radwaste ventilation system exhaust to the unit 2 vent stack (monitored by 2PR28J), not the U-1 vent stack.
Choice D is incorrect, the FHB area monitor would detect radiation in leaks around the fuel pool and the adjacent truck bay to the pool..

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:	11116
User-Defined ID:	RE10059-N01
Cross Reference Number:	
Topic:	RE10059-N01
Num Field 1:	2.7
Num Field 2:	2.7
Text Field:	E/APE059 K2.02
Comments:	Braidwood NRC Exam 2009 Question # 59 059 Accidental Liquid Radwaste Release Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: Radioactive-gas monitors TIER: 1 GROUP: 2 Cog Level: Low Task No: R-AR-002 Obj No: 4C.AR-02 Cross Ref: 10CFR55.41(b) (13)

General Data	
Technical Reference with Revision Number:	BwOP WX-277 rev. 9 P & ID M-48 sht. 54, M-78 sht. 1, M-114 sht. 2
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

60 **ID: RE10067-N01** **Points: 1.00**

Given:
- Maintenance is performing surveillance testing of fire (smoke) detectors on Unit 1.
- One detector has failed to actuate and maintenance has requested to remove the detector from its base so it can be repaired/tested at the EM shop.
- Currently all alarms are clear on panel 1PM09J.

When the detector is removed, which is the correct alarm the MCR should receive in the associated 1PM09J detection zone?

A. TROUBLE

B. TROUBLE WIRE OPEN

C. FIRE

D. FIRE WIRE OPEN

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to monitor the bypass of a fire zone detector.
When smoke detectors are removed (bypassed), the detector is removed from its base which will open the supervisory circuit between the fire detector and the fire detection control cabinet in the Aux Elec Equip Room (AEER). The result is a TROUBLE alarm in the associated zone on 1PM01J MCR panel.

Choice A is correct, see explanation above.
Choice B is incorrect, a TROUBLE WIRE OPEN alarm results when the trouble alarm circuit is opened between the AEER and the MCR TROUBLE alarm annunciator.
Choice C is incorrect, a FIRE alarm results when the detector is actuated, not when it is removed.
Choice D is incorrect, a FIRE WIRE OPEN alarm results when the fire alarm circuit is opened between the AEER and the MCR FIRE alarm annunciator.

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:	11117
User-Defined ID:	RE10067-N01
Cross Reference Number:	
Topic:	RE10067-N01
Num Field 1:	2.5
Num Field 2:	2.8
Text Field:	E/APE067 A1.03
Comments:	Braidwood NRC Exam 2009 Question # 60 067 Plant Fire On Site Ability to operate and / or monitor the following as they apply to the Plant Fire on Site: Bypass of a fire zone detector TIER: 1 GROUP: 2 Cog Level: Low Task No: R-FP-001 Obj No: 4C.FP-01 Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	BwOP FP-49 rev. 1 Big Note FP-2 rev. 7 BwHS 4009-012 rev. 4
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

61 **ID: RE10069-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power.
- Excess Letdown is in service due to normal letdown being unavailable.
- A Unit one containment release is in progress.
- All other systems are normally aligned.

The following event occurs:
- A reactor trip and SI.
- The crew is performing 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION.

During steps 8 and 9, checking Group 3 CNMT isol monitor lights and Group 6 CNMT VENT isol monitor lights, the following lights are NOT lit.

- 1CC9437B, CC FROM EXC LTDWN HX ISOL VLV
- 1CV8100, SEAL WTR RETURN CNMT ISOL VLV
- 1VQ005C, MINI-FLOW PRG EXH OUTSIDE ISOL VLV
- 1IA065, INST AIR HDR OUTSIDE CNMT ISOL VLV

- All OTHER Group 3 CNMT isol and Group 6 CNMT VENT isol lights are LIT.

With the above conditions, which of the following systems has a containment penetration that is NOT isolated?

A. Component Cooling Water (CC)

B. Seal Water Return (CV)

C. Containment Mini-Purge (VQ)

D. Instrument Air (IA)

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to monitor isolation valves as they apply to loss of containment integrity (determine which path is not isolated).
With all other valves closed (all other monitor lights lit) the examinee needs to determine which of the four containment penetrations has single valve isolation.

Choice A is correct, CC to Excess Letdown HX has single valve isolation on the inlet and outlet flowpath.
Choice B is incorrect, the CV (seal return) outlet path has also has 1CV8112 (& 1CV8113 check valve) inside containment.
Choice C is incorrect, the VQ (mini flow) outlet path also has 1VQ005A valve inside containment.

Choice D is incorrect, the IA path also has 1IA066 inside containment.

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:	11118
User-Defined ID:	RE10069-N01
Cross Reference Number:	
Topic:	RE10069-N01
Num Field 1:	3.5
Num Field 2:	3.7
Text Field:	E/APE069 A1.01
Comments:	Braidwood NRC Exam 2009 Question # 61 069 Loss of Containment Integrity Ability to operate and / or monitor the following as they apply to the Loss of Containment Integrity: Isolation valves, dampers, and electropneumatic devices TIER: 1 GROUP: 2 Cog Level: High Task No: R-OA-011 Obj No: 3S.PC-01 Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	Bwd Big Note CV-1 rev. 11 Bwd Big Note CC-2 rev. 7 Bwd Big Note VP-2 rev. 8
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

62 **ID: RE10051-N01** **Points: 1.00**

Given:
- Unit 1 reactor power is 65% and stable.
- Main Generator load is 800 MW and stable.
- 1C zone Main Condenser vacuum is 1.6 in. HgA and stable.

The following occurs:
- 1100: A Condenser air leak develops in the 1C zone causing Main Condenser pressure to RISE at a constant rate of 0.2 inches HgA/minute.

- 1109: A ramp down at 10 MW/minute is commenced in an attempt to stabilize condenser pressure.

- Field activities are ongoing to find and isolate the condenser leak.
- The NSO is directed to monitor current parameters and keep the crew informed before reaching reactor trip criteria.

With the above conditions, which of the following is the closest approximate time the unit will reach the unacceptable region of Figure 1BwOA SEC-3-1?

(assuming the Load DROP rate remains constant and the rate of Main Condenser pressure RISE remains constant throughout the event)
(Note: Figure 1BwOA SEC-3-1 is attached.)

A. 1119.

B. 1124.

C. 1127.

D. 1132.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee ability to determine conditions requiring unit trip as applied to loss of condenser vacuum.
Using 1BwOA SEC-3-1:
Plot initial conditions of 800 MW and 1.6" HgA
Pressure rises to 3.4" in first nine minutes.
Drop 10 MW and raise .2" HgA every subsequent minute and the plot will intercept the NOT ACCEPTABLE line at approx. 620 MW and 7.0" HgA pressure. 7.0" - 1.6" = 5.4" total pressure rise. 5.4"/0.2" per min = 27 min. after initiating event. 1100 + 27 = 1127.

Choice A is incorrect, 1119 is approx. time pressure would reach the 5.5" horizontal line. (trip criteria at < 480 MW).
Choice B is incorrect, 1124 is approx. time if plotted without accounting for first 9 min. of pressure rise with no ramp.
Choice C is correct, see explanation above.
Choice D is incorrect, 1132 is approx. time pressure would reach the 8.0" horizontal line. (trip criteria at > 710 MW).

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:	11119
User-Defined ID:	RE10051-N01
Cross Reference Number:	RE20051-001
Topic:	RE10051-N01
Num Field 1:	3.9
Num Field 2:	4.1
Text Field:	E/APE051 A2.02

Comments:

Braidwood NRC Exam 2009 Question # 62
051 Loss of Condenser Vacuum
Ability to determine and interpret the following as they apply to
the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip

TIER:	1
GROUP:	2
Cog Level:	High
Task No:	R-OA-088
Obj No:	3D.OA-19-C
Cross Ref:	10CFR55.41(b)(10)

General Data	
Technical Reference with Revision Number:	1BwOA SEC-3 rev. 104 figure 1BwOA SEC-3-1.
Justification for Non SRO CFR Link:	
Question History:	Previous version used in Bwd ILT COPS phase
Question Source:	Modified from Bwd ILT bank question #RE20051-001
Low KA Justification (if required):	
Revision History:	Revised question stem changing initial power, condenser pressure and time before starting ramp. Also changed question from finding reactor power when trip required to finding time trip required. (Finding reactor power based on MW at low power levels is not accurate due to secondary inefficiencies.)
ILT	
Supplied Ref (If appropriate):	ATTACH Figure 1BwOA SEC-3-1
Excluded Reference:	

63 **ID: RE1WE02-N01** **Points: 1.00**

Given the following sequence of events:
- Unit 1 is at 100% with pressurizer pressure channel, 1P-455 in test.
- ALL other equipment is in normal alignment.
- Subsequently, pressurizer pressure transmitter 1PT-458 fails low causing a low pressurizer pressure SI.
- Five minutes later, the crew has transitioned to 1BwEP ES-1.1, SI TERMINATION.
- The NSO depresses BOTH SI RESET pushbuttons.
- The BYPASS - PERMISSIVE LIGHTS silence, acknowledge, and reset pushbuttons are depressed.

With the above conditions, which of the following BYPASS - PERMISSIVE LIGHTS will be lit?

A. SI ACTUATED

B. PZR LOW PRESS SI BLOCK PERMISSIVE P11

C. PZR SI BLOCKED TRN A

D. AUTO SI BLOCKED

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee ability to verify alarms are consistent with plant conditions during SI termination.

Choice A is incorrect, SI ACTUATED will go dark after reset pushbuttons are depressed.
Choice B is incorrect, P11 will light when 2 of 3 pressurizer pressure channels is below 1930 psig, however channel 458 does not input into P11.
Choice C is incorrect, this alarm lights when prssure is <P11 and the SI lo press SI block switches are taken to block.
Choice D is correct, resetting SI later than 1 minute after an auto SI will block auto SI and light the AUTO SI BLOCKED alarm.

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:	11120
User-Defined ID:	RE1WE02-N01
Cross Reference Number:	
Topic:	RE1WE02-N01
Num Field 1:	4.2
Num Field 2:	4.2
Text Field:	W/E02 2.4.46
Comments:	Braidwood NRC Exam 2009 Question # 63 E02 SI Termination Ability to verify that the alarms are consistent with the plant conditions. TIER: 1 GROUP: 2 Cog Level: High Task No: R-EP-036 Obj No: 3D.EP-02-E Cross Ref: 10CFR55.41(b) (7)

General Data	
Technical Reference with Revision Number:	1BwEP ES-1.1 rev. 201
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

64 **ID: RE1WE09-N01** **Points: 1.00**

Given the following plant conditions on Unit 1:

- 1BwEP ES-0.2 "NATURAL CIRCULATION COOLDOWN" is in progress.
- As RCS pressure is lowered through 1300 psig, a rapid rise in PZR level is observed.
- Charging and letdown flow are approximately equal.
- The RCS cooldown rate is 25 ^{o}F/hr.

The rapid rise in PZR level indicates that...

A. natural circulation has stopped, RCS cooldown rate should be raised.

B. SI accumulators have injected, the accumulators should be isolated.

C. the reactor head is voiding, RCS pressure should be raised.

D. RCS mass is rising, excess letdown should be placed in service.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of operational implications of conditions (prz level) and remedial actions associated with a natural circ cooldown. Rapid pressurizer level changes are indication that vessel head voids are forming due to a reduction in vessel head bypass flow. 1BwEP ES-0.2 has the operator repressurize the RCS to collapse potential voids.

Choice A is incorrect, pressurizer level is not an indication used for natural circ conditions, and raising the cooldown rate would likely raise the void growth.
Choice B is incorrect, SI accumulators do not inject until pressure drops to approx 650 psig. At 1300 psig. this should not be a concern.
Choice C is correct, see explanation above.
Choice D is incorrect, a normal cooldown would have pressurizer level lowering slightly as the RCS volume shrinks.

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:	11121
User-Defined ID:	RE1WE09-N01
Cross Reference Number:	BWLC3DEP9003
Topic:	RE1WE09-N01
Num Field 1:	3.3
Num Field 2:	3.6
Text Field:	W/E09 K1.03
Comments:	Braidwood NRC Exam 2009 Question # 64 E09 Natural Circulation Operations Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation Operations): Annunciators and conditions indicating signals, and remedial actions associated with the (Natural Circulation Operations). TIER: 2 GROUP: 1 Cog Level: High Task No: R-EP-002 Obj No: 4D.EP-24 Cross Ref: 10CFR55.41(b)(10)

General Data	
Technical Reference with Revision Number:	1BwEP ES-0.2 rev. 201
Justification for Non SRO CFR Link:	
Question History:	Bwd LORT bank
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None

65 **ID: RE1WE13-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power, normal alignment.
- A loss of off site power occurs (LOOP).
- All equipment operates as designed.

Two minutes after the LOOP, the NSO monitoring 1PM04J notes two SG PORV controllers with abnormal indications.

The NSO then depresses a pushbutton on both the 1A and 1D SG PORV controllers because...

A. the automatic function is currently NOT available and the controllers need to be placed in MANUAL for SG over pressure protection.

B. an integral function has accumulated in the controllers and the signal needs to be cleared for proper SG PORV operation.

C. the controllers are currently de-energized and need to be placed in manual to enable LOCAL SG PORV operation.

D. the controllers have defaulted to manual mode and the controllers need to be placed in AUTO for SG over pressure protection.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of
reason for manipulation of controls required to obtain desired results to prevent steam generator over pressurization.
Upon loss of power to a SG PORV controller, the M/A station output fails low (closed) and when power is restored the M/A station returns in the manual operating mode. Therefore, the Auto pushbutton must be depressed to return auto over pressure protection. The power to the 1A & 1D controllers (1PA33J) is auctioneered from MCC 131X1 or 131X2, which both momentarily loss power during the LOOP event. The 1B and 1C PORV controllers (1PA34J) are auctioneered powered from 132X1 or Inst. bus 114. Since the inst. bus never loses power during the event, the 1B and 1C PORV controllers do not loss power and remain in auto.

Choice A is incorrect, the auto function is available if all equipment operates as desired (as stated in question stem). The distractor is credible because placing controllers in manual that have failed auto function or input to auto function is common practice for MCR operator.
Choice B is incorrect, if the plant operated as designed (as stated in question stem) a LOOP will not cause a reactor trip or SG over pressure condition, so there is no reason for the PORV controllers to have any integral accumulation. The distractor is credible because cycling controllers to manual to clear accumulated integral is common practice for a MCR operator, but typically it is something that is done to level controllers when restoring to auto level control.
Choice C is incorrect, if the plant operated as designed (as stated in question stem) the controller power is restored when the ESF bus power is restored from the DG. The distractor is credible because MCR operator action to de-energize the PORV motors (with the PORV C/S) is required to enable local PORV operation.

Choice D is correct, see explanation above.

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:	11122
User-Defined ID:	RE1WE13-N01
Cross Reference Number:	
Topic:	RE1WE13-N01
Num Field 1:	3.2
Num Field 2:	3.4
Text Field:	W/E13 K3.03
Comments:	Braidwood NRC Exam 2009 Question # 65 E13 Steam Generator Overpressure Knowledge of the reasons for the following responses as they apply to the (Steam Generator Overpressure): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations. TIER: 1 GROUP: 2 Cog Level: Low Task No: R-MS-002 Obj No: 3C.MS-06-A Cross Ref: 10CFR55.41(b)(7)

General Data	
Technical Reference with Revision Number:	Big note MS-7 rev. 5
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

66 **ID: RG10018-N01** **Points: 1.00**

In accordance with OP-AA-111-101 “Operating Narrative Logs and Records”, if an erroneous log entry is on a hand written or printed record, the correction should be made by...

A. completely blacking out the erroneous entry and inserting corrected information.

B. drawing a single line through the erroneous entry and inserting corrected information.

C. discarding the erroneous record and recreating a correct record.

D. writing "DELETE" in the left hand column next to the erroneous entry and inserting corrected information.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to make accurate and concise logs.
OP-AA-111-101, “Operating Narrative Logs and Records”, Section 4.1.10 specifically directs drawing a single line through erroneous log entries.

Choice A is incorrect, see explanation above.
Choice B is correct, see explanation above.
Choice C is incorrect, see exlpanation above.
Choice D is incorrect, see explanation above.

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:	11123
User-Defined ID:	RG10018-N01
Cross Reference Number:	
Topic:	RG10018-N01
Num Field 1:	3.6
Num Field 2:	3.8
Text Field:	GEN 2.1.18
Comments:	Braidwood NRC Exam 2009 Question # 66 Ability to make accurate, clear, and concise logs, records, status boards, and reports. TIER: 3 CATEGORY: 1 Cog Level: Low Task No: R-AM-006 Obj No: 3E.AM-038-A Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	OP-AA-111-101 rev. 7
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

67 **ID: RG10040-N01** **Points: 1.00**

In accordance with OU-AP-200, ADMINISTRATIVE CONTROLS DURING FUEL HANDLING ACTIVITIES FOR BYRON AND BRAIDWOOD, which of the following requires suspension of fuel assembly movements in containment?

A. Loss of one Gamma Metrics source range post accident neutron monitor.

B. Refueling cavity boron concentration drops by 25 ppm from yesterday's sample to today's sample.

C. Sustained wind in the area of 35 mph.

D. Refueling cavity water level is 23' 6" above the top of the Reactor Vessel Flange.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of refueling administrative requirements.
OU-AP-200 section 4.5.1 and 4.8.9 list conditions that require suspension of core alterations. All choices are parameters that would be typically reported to or monitored by the NSO.

Choice A is incorrect, limit is less than 2 total source range neutron monitors, however still have N-31, N-32 and another PANM.
Choice B is correct, limit is 20 ppm change maximum.
Choice C is incorrect, limit is 40 mph maximum.
Choice D is incorrect, limit is 23' minimum.

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:	11124
User-Defined ID:	RG10040-N01
Cross Reference Number:	
Topic:	RG10040-N01
Num Field 1:	2.8
Num Field 2:	3.9
Text Field:	GEN 2.1.40
Comments:	Braidwood NRC Exam 2009 Question # 67 Knowledge of refueling administrative requirements. TIER: 3 CATEGORY: 1 Cog Level: Low Task No: R-AM-009 Obj No: 4E.AM-28 Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	OU-AP-200 rev.11
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

68 **ID: RG10044-N01** **Points: 1.00**

Given:
- Unit 1 is in a refueling outage.
- Fuel movements began 24 hours ago and are currently in progress.
- At 1700 hours, the Fuel Handling Supervisor calls the Control Room and requests that the operating RH pump be stopped to allow fuel movement near the hot leg nozzle.
- NO other activities affecting the RCS are in progress.

Upon reviewing the unit logs, the NSO determines the following:

- The RH pump was stopped at 0700 for 15 minutes to allow fuel movement.

- The RH pump was stopped again at 1300 for 30 minutes for the same purpose.

At 1700 hours, per Tech Specs, the maximum time the RH pump can be secured without violating RHR loop requirements is...

A. 0 minutes.

B. an additional 15 minutes

C. an additional 30 minutes.

D. an additional 60 minutes

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of systems operated from the MCR in support of fuel operations.
TS LCO 3.9.5 allows securing required RHR loop for $\leq$ 1 hour per 8 hour period provided no ops are permitted that would cause reduction of RCS boron concentration. Since no other ops are in progress and the RH pump has been secured only 30 minutes in the past 8 hours, an additional 30 minutes is allowed.

Choice A is incorrect, would be correct if note was not in TS 3.9.5.
Choice B is incorrect, would be correct if note was 1 hour in past 12 (surv. frequency of TS 3.9.5).
Choice C is correct, see explanation above.
Choice D is incorrect, would be correct if 60 minute shut down of RH was allowed any time without 8 hour restriction.

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:	11126
User-Defined ID:	RG10044-N01
Cross Reference Number:	BWLI-AP03102
Topic:	RG10044-N01
Num Field 1:	3.9
Num Field 2:	3.8
Text Field:	GEN 2.1.44
Comments:	Braidwood NRC Exam 2009 Question # 68 Knowledge of RO duties in the control room during fuel handling, such as | responding to alarms from the fuel handling area, communication with the fuel | storage facility, systems operated from the control room in support of fueling | operations, and supporting instrumentation. TIER: 3 Category: 1 Cog Level: High Task No: R-RH-004 Obj No: 3C.RH-03-G Cross Ref: 10CFR55.41(b)(10)

General Data	
Technical Reference with Revision Number:	Bwd Tech Spec 3.9.5
Justification for Non SRO CFR Link:	
Question History:	Not currently used in ILT
Question Source:	Bwd ILT Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

69 **ID: RG20036-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power, normal alignment.
- Maintenance is performing 4 KV and 6.9 KV breaker cubicle inspections.
- The inspections require each breaker to be racked out, inspected/cleaned, then racked back in.

Which one, of the following normally open breakers, would require an LCO 3.8.1, AC SOURCES-OPERATING, entry during the inspection?

A. ACB 1411

B. ACB 1414

C. ACB 1432

D. ACB 1572

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to analyze effect of maintenance activities, such as degraded power sources, on the status of LCOs.
Per tech spec 3.8.1, two "qualified" circuits per bus between the offsite transmission network and the onsite Class 1E AC Electrical Power Distribution System are required (i.e. two ways to power 4.16 KV ESF buses from offsite).
The qualified circuits are normal feeds from SATs and
Cross-ties from others unit's SATs through their respective ESF bus.(Examinees are expected to recognize the breaker function solely from the numbering scheme).

Choice A is incorrect, 1411 is bus 141 cross tie feed to bus 143. LCO 3.4.9 would be required for pressurizer heaters capability if 1411 was racked out, but not LCO 3.8.1.
Choice B is correct, Racking out 1414 would remove the cross tie (alternate) 4 KV ESF bus source for both units requiring 3.8.1 entry on both units.
Choice C is incorrect, 1432 is SAT feed to non-ESF bus 143. No LCO is required.
Choice D is correct, 1572 is SAT feed to non-ESF bus 157. No LCO is required.

Question 69 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11127
User-Defined ID:	RG20036-N01
Cross Reference Number:	
Topic:	RG20036-N01
Num Field 1:	3.1
Num Field 2:	4.2
Text Field:	GEN 2.2.36
Comments:	Braidwood NRC Exam 2009 Question # 69 Ability to analyze the effect of maintenance activities, such as degraded power | sources, on the status of limiting conditions for operations. TIER: 3 CATEGORY: 2 Cog Level: High Task No: R-AP-001 Obj No: 3C.AP-01-B Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	Tech Spec 3.8.1
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

70 **ID: RG20044-N01** **Points: 1.00**

Given the following plant conditions on Unit 1:

- Plant conditions have been stable for the past 15 minutes.
- Loop 1A T_{AVE} is 572.0 °F.
- Loop 1B T_{AVE} is 570.0 °F.
- Loop 1C T_{AVE} is 570.0 °F.
- Loop 1D T_{AVE} is 568.0 °F.
- T_{REF} is 568.0 °F.
- The Rod Bank Select Switch is in MANUAL.

If the Rod Bank Select Switch is placed in AUTO, the control rods would initially ...

A. NOT step.

B. step IN at 8 steps/minute.

C. step IN at 40 steps/minute.

D. step IN at 54 steps/minute.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee ability to interpret MCR indications (Tave/Tref) to verify status of rod control system (demand) and understand affect of taking rods to auto.
Rod control unit uses auctioneered high Tave channel and compares it to Tref to calculate demand. Also uses rate of change between auctioneered high NI channel and P-imp, however with stable conditions (given in stem) this input can be ignored. Rods will step at 8 steps per/min with a 1.5°F to 3°F mismatch. From 3°F to 5°F the rate will rise linearly from 8 to 72 steps per/min. The question step has a 4°F mismatch which interpolates to 40 steps per/min.

Choice A is incorrect, would be correct if rod control used auctioneered low Tave.
Choice B is incorrect, would be correct if rod control used average Tave.
Choice C is correct, see explanation above
Choice D is incorrect, would be correct if calculated with linear rise from 1.5°F to 5°F

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:	11128
User-Defined ID:	RG20044-N01
Cross Reference Number:	BWLC3DOA1021
Topic:	RG20044-N01
Num Field 1:	4.2
Num Field 2:	4.0
Text Field:	GEN 2.2.44
Comments:	Braidwood NRC Exam 2009 Question # 70 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. TIER: 3 CATEGORY: 2 Cog Level: High Task No: R-RD-001 Obj No: 3C.RD-02-A Cross Ref: 10CFR55.41(b) (6)

General Data	
Technical Reference with Revision Number:	I1-RD-XL-01 rev 3a Bwd ILT Rod Control lesson plan
Justification for Non SRO CFR Link:	
Question History:	Bwd LORT Bank
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

71 **ID: RG30004-N01** **Points: 1.00**

The following conditions exist:

- A General Emergency has been declared due to a Large Break LOCA.

- A High Radiation area needs to be accessed to PROTECT VALUABLE PROPERTY.

- An Emergency Responder has volunteered to perform the job.
- The volunteer has a current year to date exposure of 3 Rem.

Which one of the following describes the MAXIMUM dose the emergency responder is allowed to receive for this activity?

A. 7 REM TEDE

B. 10 REM TEDE

C. 22 REM TEDE

D. 25 REM TEDE

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of radiation exposure limits under emergency conditions.

Choice A is incorrect, this is the dose allowed for protecting valuable property minus his current dose.
Choice B is correct, this is the dose allowed for protecting valuable property per RP-AA-203 section 4.5.3. Exposure limit is "per activity" and current dose is not subtracted from the limit
Choice C is incorrect, this is the dose for saving a life minus his current dose
Choice D is incorrect, this is the allowed dose for lifesaving activity.

Question 71 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:	11131
User-Defined ID:	RG30004-N01
Cross Reference Number:	
Topic:	RG30004-N01
Num Field 1:	3.2
Num Field 2:	3.7
Text Field:	GEN 2.3.04
Comments:	Braidwood NRC Exam 2009 Question #71 Knowledge of radiation exposure limits under normal or emergency conditions. TIER: 3 CATEGORY: 3 Cog Level: Low Task No: R-AM-133 Obj No: 3E.AM-133 Cross Ref: 10CFR55.41(b)(12)

General Data	
Technical Reference with Revision Number:	RP-AA-203 rev.3
Justification for Non SRO CFR Link:	
Question History:	From VC Summer previous NRC exam bank question (Focus Learning ILT exam bank ID # 24406)
Question Source:	Modified
Low KA Justification (if required):	
Revision History:	Changed from life saving event to protection of valuable property which changes answer from 25 REM to 10 REM.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

72 **ID: RG30011-N01** **Points: 1.00**

Given:
- A Unit 2 Containment Release is in progress per BwRP 6110-13T1, CONTAINMENT RELEASE FORM.

Currently, the RUNNING Aux Bldg Vent Stacks exhaust fans are:

- 0VA02CA, VA EXH FAN 0A TRN 0A
- 0VF01CA, FILTERED TANK VENTS EXH FAN
- 0VL02CA, LAB HVAC FUME HOOD EXH FAN 0A
- 0VW03CA, RADWASTE BLDG EXH FAN 0A

- ALL other Aux Bldg Vent Stacks exhaust fans and charcoal booster fans are shutdown.

With the above condition, which one of the following fan trips would require the operator to secure the Unit 2 Containment Release?

(Consider each choice separately and do NOT assume ANY standby fans are started.)

A. 0VA02CA trips.

B. 0VF01CA trips.

C. 0VL02CA trips.

D. 0VW03CA trips.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee ability to control a containment release when conditions for the release are no longer met.
When a containment release is in progress, BwRP 6110-13T1 requires that sufficient exhaust fan flow is in operation for the unit vent stack that is handling the release. There are three fan configurations that meet this requirement:
1. at least 1 VA exh fan
2. 2 Aux bldg charcoal booster fans AND 1 FH bldg charcoal booster fan
3. one lab HVAC fan (VL) for unit 1 OR one radwaste bldg fan (VW) for unit 2

The filtered vent (VF) fan discharges to unit 2 vent stack, however it is not of sufficient size to alone satisfy the flow requirements.

Choice A is incorrect, 0VA02CA discharges to U-1 stack, therefore would not affect U-2 release.
Choice B is incorrect, VF fan trip would still leave VW fan discharging to U-2 stack.
Choice C is incorrect, 0VL02CA discharges to U-1 stack, therefore would not affect U-2 release.
Choice D is correct, VW fan trip would leave only the (VF) fan discharging to the U-2 stack, which is not sufficient for a containment release.

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:	11130
User-Defined ID:	RG30011-N01
Cross Reference Number:	
Topic:	RG30011-N01
Num Field 1:	3.8
Num Field 2:	4.3
Text Field:	GEN 2.3.11
Comments:	Braidwood NRC Exam 2009 Question # 72 Ability to control radiation releases. TIER: 3 CATEGORY: 3 Cog Level: High Task No: R-VQ-004 Obj No: 4C.VQ-01 Cross Ref: 10CFR55.41(b) (13)

General Data	
Technical Reference with Revision Number:	BwRP 6110-13T1 rev. 18
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

73 **ID: RG30015-N01** **Points: 1.00**

Concerning Radiation Monitors with interlock functions:

Which one of the following RM-11 icon conditions will actuate the interlock function?

A. Magenta, COMMUNICATIONS

B. Dark Blue, OPERATE FAILURE

C. Yellow, CHANNEL ALERT ALARM

D. Half Cyan (grey), CONTROL FUNCTION

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of rad monitoring systems and alarms.

Choice A is incorrect, a communication error is a loss of data transmission between the RM-11 and the associated monitors RM-80, but interlocks will not happen.
Choice B is correct, operate failures cause interlock functions to actuate since the skid is no longer functioning to monitor the process stream.
Choice C is incorrect, alert alarm indicates rad levels are rising above normal, but interlocks do not happen until rad levels reach red, high alarm level.
Choice D is incorrect, the interlock function is bypassed when the rad skid is in the control function mode.

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:	11132
User-Defined ID:	RG30015-N01
Cross Reference Number:	
Topic:	RG30015-N01
Num Field 1:	2.9
Num Field 2:	3.1
Text Field:	GEN 2.3.15
Comments:	Braidwood NRC Exam 2009 Question # 73 Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. TIER: 3 CATEGORY: 3 Cog Level: Low Task No: R-AR-002 Obj No: 4C.AR-02 Cross Ref: 10CFR55.41(b) (11)

General Data	
Technical Reference with Revision Number:	ILT lesson plan I1-AR-XL-01 rev. 4a
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

74 **ID: RG40016-N01** **Points: 1.00**

While transitioning to 1BwEP-1, "Loss of Reactor or Secondary Coolant," after completing the actions of 1BwEP-0, "Reactor Trip or Safety Injection," an orange path is identified under SUBCRITICALITY.

With regard to implementing 1BwFR-S.1, "Response to Nuclear Power Generation/ATWS", procedure usage rules require the crew to...

A. continue current pass through the status trees, if NO red condition is encountered, then implement 1BwFR-S.1.

B. complete the actions of 1BwEP-1, then implement 1BwFR-S.1.

C. immediately implement 1BwFR-S.1, then continue current pass through the status trees.

D. implement 1BwFR-S.1 at the discretion of the Shift Manager.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of EOP implementation coordination.
BwAP 340-1, section C.4.d.4) states "An Orange condition does not require immediate action. The current pass through the status trees is to be completed, with the status of each Critical Safety Function noted. If no Red condition is encountered during the scan, then the highest priority Orange is addressed first, requiring departure from the procedure in effect. Once the actions of the _BwFR are completed assuming no Red condition has appeared, the next highest priority Orange can be addressed. The question is RO level because it is ensuring compliance with an administrative procedure (RO and SRO task) and it is NOT asking the RO to assess conditions and select a procedure (which would be SRO only task).

Choice A is correct, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

Question 74 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11134
User-Defined ID:	RG40016-N01
Cross Reference Number:	BWLI-FRS-035
Topic:	RG40016-N01
Num Field 1:	3.5
Num Field 2:	4.4
Text Field:	GEN 2.4.16
Comments:	Braidwood NRC Exam 2009 Question # 74 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. TIER: 3 CATEGORY: 4 Cog Level: Low Task No: R-AM-133 Obj No: 3E.AM-133 Cross Ref: 10CFR55.41(b)(10)

General Data	
Technical Reference with Revision Number:	BwAP 340-1 rev 24
Justification for Non SRO CFR Link:	
Question History:	Not currently used in Bwd ILT
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

75 **ID: RG40027-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power, normal alignment, when a fire occurs in the plant.
- MCR annunciator 0-37-A4, UNIT 1 AREA FIRE, alarms.
- The assist NSO is performing the Immediate Operator Actions per BwAR 0-37-A4.
- Personnel have been dispatched to investigate and extinguish the fire.
- The fire is determined to be in a specific zone and of sufficient magnitude to prompt further Immediate Operator Actions.

Per BwAR 0-37-A4, the further Immediate Operator Actions are intended to prevent...

A. loss of Diesel Generators' control power.

B. inadvertent Reactor Head Vent Valves opening.

C. inadvertent Containment Spray actuation.

D. loss of Charging Pumps suction source.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of "fire in the plant" procedures.

Choice A is incorrect, DG control power is isolated in 1BwOA ELEC-3 when attempting local start of a DG.
Choice B is incorrect, Reactor head vents are disabled in 1BwOA PRI-5 when MCR is evacuated.
Choice C is incorrect, CS pump breakers are disabled in 1BwOA PRI-5 when MCR is evacuated.
Choice D is correct, BwAR 0-37-A4 further immediate operator actions directs letdown and charging to be isolated, CV pp suction valves to be de-energized and then charging and letdown to be re-established. This is to prevent the fire from causing a short and inadvertent closing of 1CV112B or C.

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:	11135
User-Defined ID:	RG40027-N01
Cross Reference Number:	
Topic:	RG40027-N01
Num Field 1:	3.4
Num Field 2:	3.9
Text Field:	GEN 2.4.27
Comments:	Braidwood NRC Exam 2009 Question # 75 Knowledge of “fire in the plant” procedures. TIER: 3 CATEGORY: 4 Cog Level: Low Task No: R-AM-040 Obj No: 3D.AM-01-A Cross Ref: 10CFR55.41(b)(10)

General Data	
Technical Reference with Revision Number:	BwAR 0-37-A4 rev. 13 ILT lesson plan I1-FP-XL-01 rev. 7a
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

76 **ID: SS20004-N01** **Points: 1.00**

Given:
- Unit 1 is at 90% power, all systems normally aligned for the current power level.

The following occurs:
- A failure of 1CC130A controller causes letdown temperature to drop to 75°F.

Based on the above conditions and assuming NO operator actions are taken, the plant response will be _____(1)_____, AND the correct procedure the US will enter to mitigate the event is _____(2)_____.

A. (1) Control Rods will step IN
(2) 1BwOA PRI-12 UNCONTROLLED DILUTION

B. (1) 1CV129, DEMIN HI TEMP LTDWN DIVERT VLV will OPEN
(2) 1BwOA PRI-6 COMPONENT COOLING MALFUNCTION

C. (1) Control Rods will step OUT
(2) 1BwOA PRI-6 COMPONENT COOLING MALFUNCTION

D. (1) Reactor Power will RISE above 100%
(2) 1BwOA PRI-16 RESPONSE TO OVERPOWER CONDITION

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to predict impact of malfunction on CVCS (letdown temperature control) and use procedures to mitigate the consequences. The question is SRO level because it requires assessment of conditions and selection of appropriate procedure.
At a lower temperature, the letdown mixed bed demins will remove boron from the flowpath and dilute the VCT (and ultimately the RCS). The dilution will cause RCS Tave to rise and control rods will move in to compensate.

Choice A is correct, 1BwOA PRI-12, step 3, will check for low letdown temperature and then manually bypass the demins by placing 1CV129 to the VCT position. This will stop the dilution.
Choice B is incorrect, Rods will not step out even though letdown temperature is lower. The affect of dilution will cause RCS temp to rise. 1BwOA PRI-6 checks for high letdown temp, but not low letdown temperature, therefore does not mitigate the event.
Choice C is incorrect, 1CV129 auto diverts on high letdown temperature to protect the demin resin from high temps. Also 1BwOA PRI-6 checks for high letdown temp, but not low letdown temperature, therefore does not mitigate the event.
Choice D is incorrect, Reactor power will be minimally affected (will initially drop slightly as RCS temperature rises. At 90% initial power level, overpower will not be a concern as it would be at 100% initial conditions.

Question 76 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11136
User-Defined ID:	SS20004-N01
Cross Reference Number:	BWLI-PRI2044
Topic:	SS20004-N01
Num Field 1:	3.8
Num Field 2:	4.3
Text Field:	SYS004 A2.25

Comments:

Braidwood NRC Exam 2009 Question # 76
004 Chemical and Volume Control System
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: Uncontrolled boration or dilution

TIER:	2
GROUP:	1
Cog Level:	High
Task No:	S-OA-065
Obj No:	7D.OA-021-A
Cross Ref:	10CFR55.43(b)(5)

General Data	
Technical Reference with Revision Number:	Bwd ILT lesson plan I1-CV-XL-01 rev. 7c 1BwOA PRI-12 rev. 101
Justification for Non SRO CFR Link:	
Question History:	Previous version used in NOPS phase of ILT
Question Source:	Modified from Bwd ILT bank #BWLI-PRI2044
Low KA Justification (if required):	
Revision History:	Modified question stem from 1CC130A inst. air failure to a failure causing letdown temp drop. Added procedure entry to answer to meet KA and SRO level question.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

77 **ID: SS20006-N01** **Points: 1.00**

Given the following sequence of events:
- Unit 2 is at 100% power, normal alignment.
- An INADVERTENT SI occurs.
- One minute later, the ECCS high head injection header ruptures BETWEEN 2SI8801A, CHG PMPS TO COLD LEG INJ ISOL VLV and containment.
- NO other leaks exist.

Current plant conditions are:
- 2FI-917 HIGH HEAD SI FLOW is pegged HIGH.
- Pressurizer level is 28% and STABLE.
- Pressurizer pressure is 2100 psig and slowly RISING.
- RCS temperature is 557°F and STABLE.
- The crew has just completed the Immediate Actions steps of 2BwEP-0, REACTOR TRIP OR SAFETY INJECTION.

With the above conditions and in accordance with proper procedural usage, the leak will be isolated by MCR actions performed in...

A. 2BwEP ES-1.1, SI TERMINATION.

B. 2BwOA PRI-1, EXCESSIVE PRIMARY PLANT LEAKAGE.

C. 2BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

D. 2BwCA-1.2, LOCA OUTSIDE CONTAINMENT.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee ability to use procedures to mitigate consequences of ECCS header rupture.
The question is SRO level because it requires assessment of conditions and selection of appropriate procedure.
With the given plant parameters, the first procedural directed transition from 2BwEP-0 is to 2BwEP ES-1.1. Steps 1 thru 6 of 2BwEP ES-1.1 will terminate hi head SI flow and close 2SI8801A & B which will isolate the leak. (leak will be isolated from RCS by check valves inside containment.)

Choice A is correct, see explanation above.
Choice B is incorrect, PRI-1 is written assuming operation in mode 1, 2 or 3 (>P11) and no SI has occurred. The procedure diagnostic steps do not work properly with an active SI. Additionally, there is no procedural directed transitions to PRI-1 from 2BwEP-0.
Choice C is incorrect, transition to 2BwEP-1 from 2BwEP-0 is based upon cmnt rads, cmnt pressure or cmnt sump level. All three of these parameters are normal given there are no leaks inside containment.
Choice D is incorrect, transition to 2BwCA-1.2 from 2BwEP-0 is at step 32 however, it is based upon high aux bldg. rads and an evaluation that the RCS is leaking outside containment. With given conditions, only RWST water is being leaked into aux bldg. therefor, rads should be normal and leakage cannot be evaluated as RCS due to normal RCS parameters.

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:	11137
User-Defined ID:	SS20006-N01
Cross Reference Number:	
Topic:	SS20006-N01
Num Field 1:	4.0
Num Field 2:	4.4
Text Field:	SYS006 A2.11
Comments:	Braidwood NRC Exam 2009 Question # 77 006 Emergency Core Cooling System (ECCS) Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture of ECCS header TIER: 2 GROUP: 1 Cog Level: High Task No: S-EP-036 Obj No: 7D.EP-002-E Cross Ref: 10CFR55.43(b)(5) Question Difficulty: ?

General Data	
Technical Reference with Revision Number:	2BwEP-0 rev. 203 2BwEP ES-1.1 rev. 201
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

78 **ID: SS20076-N01** **Points: 1.00**

Given:
- Both Units are at 100% power, normally aligned.
- 1A SX Pump AND 2B SX Pump are RUNNING.

Unusually warm weather has caused the following elevated temperatures:
- 1A SX pp disch temp = 102.0°F
- 2B SX pp disch temp = 99.0°F

Concerning the following LCOs:
3.7.8 SX SYSTEM
3.7.9 ULTIMATE HEAT SINK

With the above conditions, the Tech Spec LCO entry(ies) required is(are) _____(1)_____ because _____(2)_____.

A. (1) ONLY 3.7.8 on Unit 1 ONLY
(2) the SX train may not be sufficient to remove decay heat loads.

B. (1) ONLY 3.7.8 on BOTH units
(2) the SX trains may not be sufficient to remove decay heat loads.

C. (1) 3.7.8 for Unit 1 ONLY AND 3.7.9 for Unit 1 ONLY
(2) the containment air temperature may exceed the peak assumed temp in the accident analysis.

D. (1) ONLY 3.7.9 for BOTH units
(2) the SX system loads may exceed maximum design temperatures within 30 days of a DBA LOCA.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee ability to recognize system parameters that are entry level conditions for tech specs.
The question is SRO level because it requires knowledge of Tech Spec bases.
Per Tech Spec 3.7.9 and bases the UHS average water temp must be ≤ 100°F. The bases for this is the to allow SX system operation for 30 days following design based LOCA.

Choice A is incorrect, 3.7.8 is the LCO for SX, not the UHS and bases is incorrect.
Choice B is incorrect, 3.7.8 is the LCO for SX, not the UHS
Choice C is incorrect, 3.7.8 is the LCO for SX, not the UHS
Choice D is correct, see explanation above.

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:	11138
User-Defined ID:	SS20076-N01
Cross Reference Number:	
Topic:	SS20076-N01
Num Field 1:	3.9
Num Field 2:	4.6
Text Field:	SYS076 2.2.42
Comments:	Braidwood NRC Exam 2009 Question # 78 076 Service Water System (SWS) Ability to recognize system parameters that are entry-level conditions for | Technical Specifications. TIER: 2 GROUP: 1 Cog Level: High Task No: S-TS-006 Obj No: 7E.TS-004 Cross Ref: 10CFR55.43(b) (2)

General Data	
Technical Reference with Revision Number:	1BwOSR 0.1-1,2,3 rev. 51 Tech Spec 3.7.9. bases
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

79 **ID: SS20013-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power.

- Instrument Bus 114 is faulted.

- An overcurrent relay problem has caused breaker 1415X, Bus 141 Feed to Xfmr 131X, to trip open and bus 131X CANNOT be reenergized.

The following subsequently occurred:
- A large break RCS LOCA.
- Containment pressure is 26 psig.
- The crew has just completed the Immediate Actions steps of 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION.
- NO other operator actions have been taken.

Currently, with the above conditions, the US will...

A. continue in 1BwEP-0, because there is at least one CS pump running.

B. transition to 1BwFR-Z.1 RESPONSE TO HIGH CONTAINMENT PRESSURE, because there are NO CS pumps running.

C. perform 1BwOA ELEC-2 LOSS OF INSTRUMENT BUS, while continuing with 1BwEP-0, even though one CS pump is running.

D. continue in 1BwEP-0, even though there are NO CS pumps running.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee ability to predict impact of loss of instrument bus on ESFAS (CS) and use procedures to correct the consequences.
The question is SRO level because it requires assessment of conditions and selection of appropriate procedure.
The loss of instrument bus 114 will deenergize the output cabinet of B train SSPS, preventing the 1B CS train from starting. The bus 131X loss will prevent 1CS019A valve from opening, which is an interlock for the A train CS pump auto start. Therefore, neither CS pump will be operating.

Choice A is incorrect, no CS pump will be operating (see explanation above).
Choice B is incorrect, the status tree monitoring does not take affect until after the crew has transitioned out of 1BwEP-0. Therefore, transition to 1BwFR-Z.1 is not appropriate at this time even though orange conditions are present for containment.
Choice C is incorrect, 1BwOA ELEC-2 is not specified anywhere in 1BwEP-0 to use in conjunction. Additionally, 1BwOA ELEC-2 attempts to restore the instrument bus, which cannot be done when the bus is faulted. Also, no CS pumps will be running.
Choice D is correct, 1BwEP-0, step 14, checks status of CS system, manually actuates CS from the 2 of 2 MCB switches and then manually aligns the CS valves to allow pump start in Attachment C.

Question 79 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11139
User-Defined ID:	SS20013-N01
Cross Reference Number:	BWLC3DCS1004
Topic:	SS20013-N01
Num Field 1:	3.6
Num Field 2:	4.2
Text Field:	SYS013 A2.04
Comments:	Braidwood NRC Exam 2009 Question # 79 013 Engineered Safety Features Actuation System (ESFAS) Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; Loss of instrument bus TIER: 2 GROUP: 1 Cog Level: High Task No: R-EF-001 Obj No: 3C.EF-01-B Cross Ref: 10CFR55.43(b) (5)

General Data	
Technical Reference with Revision Number:	1BwEP-0 rev. 202
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	Modified from Bwd LORT bank # BWLC3DCS1004.
Low KA Justification (if required):	
Revision History:	Changed stem of question to require procedure selection as part of answer.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

80 **ID: SS20062-N01** **Points: 1.00**

Given:
- Unit 1 has experienced a loss of all AC power.
- Unit 2 has NOT been affected.
- The 1A DG is OOS in maintenance outage mode.
- The 1B DG is running, but breaker 1423 failed to close automatically AND manually.
- The crew is performing 1BwCA-0.0 LOSS OF ALL AC POWER, at step 5, TRY TO RESTORE POWER TO ANY/BOTH UNIT 1 4KV ESF BUSES.

With the above conditions, the NEXT action the US will direct is...

A. depress the SI RECIRC SUMP ISOL VLV pushbuttons to remove a trip signal from breakers 1411 and 1421 to allow for ESF bus to non-ESF bus cross ties.

B. place the 1B DG control switch at 1PM01J in PTL to prevent the blackout sequencer from automatically loading bus 142.

C. depress the EMERGENCY STOP pushbutton on the 1B DG to remove a trip signal from breaker 1424 to allow for ESF bus 142 cross tie to Unit 2.

D. SYNC and close breaker 1424 to prepare for ESF bus 142 cross tie to Unit 2.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of operational effect of AC electrical distribution operator task performed outside the control room.
The question is SRO level because the SRO drives 1BwCA 0.0 and orders the actions of the steps to cross tie and restore power within a time critical situation.

Choice A is incorrect, during a loss of all AC power, non-ESF bus to ESF bus cross tie does not take place until the ESF bus is re-energized and the crew has transitioned into a recovery procedure.
Choice B is incorrect, the blackout sequencer is defeated by placing the ESF load control switches to PTL.
Choice C is correct, The DG Emergency Stop Pushbutton is depressed in 1BwCA-0.0 to remove a trip signal from the cross tie brk 1424. This trip signal is present if a UV signal is present on the bus and the DG is at rated speed and voltage. Therefore this trip must be removed before the future procedure cross tie can be performed.
Choice D is incorrect, closing 1424 is the last step performed to cross tie bus 142 to 242. There are several steps that need to be performed before this and the question stem asks for the NEXT action.

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:	11140
User-Defined ID:	SS20062-N01
Cross Reference Number:	
Topic:	SS20062-N01
Num Field 1:	4.2
Num Field 2:	4.1
Text Field:	SYS062 2.4.34
Comments:	Braidwood NRC Exam 2009 Question # 80 062 AC Electrical Distribution System Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. TIER: 2 GROUP: 1 Cog Level: High Task No: S-CA-005 Obj No: 8D.CA-001-E Cross Ref: 10CFR55.43(b)(5)

General Data	
Technical Reference with Revision Number:	1BwCA-0.0 rev 201 Bwd ILT lesson plan I1-CA-XL-01 rev. 6a
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

81 **ID: SS20034-N01** **Points: 1.00**

Given:
- New Fuel receipt activities are being conducted in compliance with applicable procedure and Tech Spec/TRM requirements.
- A cask containing new fuel is being transferred from the transport vehicle to the new fuel inspection area where another cask containing new fuel is already placed.
- During transfer, the fuel handling building crane hoist brake fails, and the cask is descending quickly.

Based on the above conditions, identify the ONE selection below that predicts the impact of the load drop?

A. The cask may damage spent fuel in the pool because the crane must travel over the pool to reach the new fuel inspection area.

B. The cask will not damage spent fuel in the pool because the crane is administratively AND mechanically/electrically prohibited from traveling over the pool.

C. The cask should not damage spent fuel in the pool because the crane is administratively ONLY (not mechanically/electrically) prohibited from traveling over the pool.

D. If the descending cask contacts the second cask, the minimum "edge to edge" distance requirement for new fuel will be violated.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to predict impact of dropped cask on fuel handling system. Question only tests part (a) of K/A IAW NUREG ES-401 section D.2.a (page 6) because part (b) does not create a discriminating question.
The question is SRO level because it requires knowledge of fuel handling facilities in the fuel handling bldg. that are not related to the MCR.

Choice A is incorrect, the new fuel inspection are is adjacent to the vehicle unloading area, therefor new fuel casks do not require travel over the pool.
Choice B is correct, TRM 3.9.d (admin) prohibits loads over 2000 pounds from traveling over the pool. Additionally, there is a mechanical interlock to prevent the crane from traveling over the pool to meet surv. requirement 3.9.d.1.
Choice C is incorrect, mechanical interlock does exist preventing crane operator from violating admin requirement.
Choice D is incorrect, Requirement for 12" minimum edge to edge distance is for new fuel assemblies that are outside their shipping containers. Since the new fuel is still in the cask and the casks are robust enough to prevent breaking open from a drop as described, there is no chance the 12" minimum edge to edge requirement will be violated.

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:	11141
User-Defined ID:	SS20034-N01
Cross Reference Number:	
Topic:	SS20034-N01
Num Field 1:	3.4
Num Field 2:	3.9
Text Field:	SYS034 A2.02

Comments:

Braidwood NRC Exam 2009 Question # 81
034 Fuel Handling Equipment System (FHES)
Ability to (a) predict the impacts of the following malfunctions or
operations on the Fuel Handling System ; and (b) based on those
predictions, use procedures to correct, control, or mitigate the
consequences of those malfunctions or operations:
Dropped cask

TIER:	2
GROUP:	2
Cog Level:	Low
Task No:	S-FH-004
Obj No:	7C.FH-003
Cross Ref:	10CFR55.43(b)(7)

General Data	
Technical Reference with Revision Number:	TRM 3.9.d Big note PC-4 rev. 6 OU-AP-200 rev. 11
Justification for Non SRO CFR Link:	
Question History:	TMI previous NRC exam
Question Source:	Modified from TMI previous NRC exam bank question (Focus Learning ILT exam bank ID # 25358)
Low KA Justification (if required):	
Revision History:	Changed answers and distractors to ask for reasons for impact on FH system.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

82 **ID: SS20035-N01** **Points: 1.00**

Given:
- A small break LOCA has occurred on Unit 1.

During the initial performance of 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, at step 15, the following plant conditions are noted:

- Containment pressure 6 psig and rising.
- CETCs indicate 820°F.
- RCS pressure is 1750 psig and stable.
- S/G pressures are 1175 psig.
- Both AF pumps failed to start and CANNOT be manually started.
- S/G levels (NR): 1A S/G 25%, 1B S/G 24%, 1C S/G 26%, 1D S/G 30%

Based on the above conditions, the NEXT procedure the US will transition to is...

A. 1BwEP-1 LOSS OF REACTOR OR SECONDARY COOLANT, because SG levels are currently adequate.

B. 1BwFR-C.2 RESPONSE TO DEGRADED CORE COOLING, because core cooling is the highest priority safety function not being met.

C. 1BwFR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK, and 1BwFR-H.1 will be performed because a secondary heat sink is necessary.

D. 1BwFR-H.1 RESPONSE TO LOSS OF SECONDARY HEAT SINK, then immediately transition to 1BwFR-C.2 RESPONSE TO DEGRADED CORE COOLING because a secondary heat sink is NOT required.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee ability to predict impact on SG operations of small break LOCA and use procedures to mitigate the consequences. The question is SRO level because it requires assessment of conditions and selection of appropriate procedure.

Choice A is incorrect, SG levels are not currently adequate with adverse containment. Requirement is 31% or greater.
Choice B is incorrect, status tree monitoring does not take affect until after the crew has transitioned out of 1BwEP-0. Therefore, transition to 1BwFR-C.2 is not appropriate at this time even though orange conditions are present for core cooling.
Choice C is correct, SG levels are not adequate with 0 AF flow, transition to FR-H.1 is directed in step 15 of EP-0. First step of FR-H.1 then will check if a heat sink is required by verifying RCS pressure is greater than SG pressures. Since RCS pressure is greater, a heat sink is required and FR-H.1 steps will be performed.
Choice D is incorrect, will not immediately transition back to EP-0. Heat sink is required. See answer explanation.

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:	11142
User-Defined ID:	SS20035-N01
Cross Reference Number:	
Topic:	SS20035-N01
Num Field 1:	4.5
Num Field 2:	4.6
Text Field:	SYS035 A2.06

Comments:

Braidwood NRC Exam 2009 Question # 82
035 Steam Generator System (S/GS)
Ability to (a) predict the impacts of the following malfunctions
or operations on the GS; and (b) based on
those predictions, use procedures to correct, control,
or mitigate the consequences of those malfunctions or
operations: Small break LOCA
(CFR: 41.5 /

TIER:	2
GROUP:	2
Cog Level:	High
Task No:	S-EP-005
Obj No:	7D.EP-001-A
Cross Ref:	10CFR55.43(b) (5)

General Data	
Technical Reference with Revision Number:	1BwEP-0 rev. 202 1BwFR H-1 rev. 200
Justification for Non SRO CFR Link:	
Question History:	Beaver Valley previous NRC exam
Question Source:	Modified from Beaver Valley previous NRC exam bank question (Focus Learning ILT exam bank ID # 29438)
Low KA Justification (if required):	
Revision History:	Changed stem conditions from large break LOCA to small break LOCA . Changed answer and two distractors.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

83 **ID: SS20072-N01** **Points: 1.00**

Given:
- Unit 1 is in MODE 4 after a normal shutdown and cooldown in accordance with all procedures.
- RCS Pressure was manually depressurized to 350 psig.
- SI Accumulators are ISOLATED.

A few minutes ago a Containment Area Rad monitor alarmed and the crew noted the following:
- PZR Level is DROPPING.
- Letdown is ISOLATED.
- Charging flow is 200 gpm.

With the above conditions, the next procedure the US will enter is _____, because it contains the specific actions to mitigate these conditions.

A. 1BwOA PRI-1 EXCESSIVE PRIMARY PLANT LEAKAGE.

B. 1BwOA S/D-2 SHUTDOWN LOCA.

C. 1BwOA PRI-4 HIGH REACTOR COOLANT ACTIVITY.

D. 1BwEP-1 LOSS OF REACTOR OR SECONDARY COOLANT.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee knowledge of abnormal condition procedures.
The question is SRO level because it requires assessment of conditions and selection of appropriate procedure.

Choice A is incorrect, PRI-1 is written for leaks in Mode 1-3 with SI accumulators not isolated. If the SRO chose to enter PRI-1, then at step 2 RNO transition would be made to S/D-2 because plant was not in the mode of applicability listed in PRI-1. Therefore PRI-1 is incorrect because the mitigation steps for this situation are not contained in it.
Choice B is correct, S/D 2 is written for modes 3 and 4 after SI accumulators are isolated. Mitigation steps of S/D-2 will be to manually restore and align ECCS, then cool down the plant and depressurize.
Choice C is incorrect, PRI-4 is applicable for the gross failed fuel monitor alarm, not a containment area monitor.
Choice D is incorrect, EP-1 would not be the next procedure entry because it is not a direct entry procedure.

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:	11143
User-Defined ID:	SS20072-N01
Cross Reference Number:	
Topic:	SS20072-N01
Num Field 1:	4.0
Num Field 2:	4.2
Text Field:	SYS072 2.4.11
Comments:	Braidwood NRC Exam 2009 Question # 83 072 Area Radiation Monitoring (ARM) System Knowledge of abnormal condition procedures. TIER: 2 GROUP: 2 Cog Level: High Task No: S-OA-131 Obj No: 7D.OA-052-A Cross Ref: 10CFR55.43(b) (5)

General Data	
Technical Reference with Revision Number:	1BwOA S/D-2 rev. 103
Justification for Non SRO CFR Link:	
Question History:	From Braidwood previous NRC exam bank question (Focus Learning ILT exam bank ID # 19845) (Question was not on previous two Bwd NRC exams)
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	Changed two distractors from 1BwOA SEC-4 and 1BwOA PRI-10 to 1BwOA PRI-4 and 1BwEP-1 to make them more plausible.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

84 **ID: SE10015-N01** **Points: 1.00**

Given:
- Unit 1 is at 400°F during a plant startup.
- 1C and 1D RCPs are in OPERATION.
- Due to unsatisfactory oil samples, 1A and 1B RCPs are OOS for maintenance.
- The Control Rod Drive MG sets are operating and the Reactor Trip Breakers are closed for testing.
- The RCS is being diluted to ECC boron concentration.

The following occurs:
- A seal leakoff problem develops with the 1C RCP and the 1C RCP is tripped.

With the above conditions, which one of the following actions must be taken?

A. Initiate action to place 1 train of RH in service immediately.

B. Initiate action to restore one RCP immediately.

C. Verify Shutdown Margin is within limits of the COLR within one hour.

D. Open the Reactor Trip Breakers within one hour.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee ability to apply Tech Specs for a system during a RCP malfunction.
The question is SRO level because it requires knowledge of Tech Spec actions with 1 hr or less completion times.
Per T.S. 3.4.5, RCS Loops Mode 3, two operable RCS loops shall be in operation when Rod Control system is capable of rod withdrawl. Tripping the 1C RCP would drop the number of operational loops to one. LCO 3.4.5 Condition A requires placing rod control in a condition incapable of rod withdrawl. Opening the reactor trip breakers would satisfy the condition.
Choice A is incorrect, see explanation above. Placing RH loop in operation would be correct if unit was in Mode 4 with no loops in operation.
Choice B is incorrect, initiate action to restore one loop is required action for LCO 3.4.5 cond. B (no loops in operation).
Choice C is incorrect, verify SDM is one hour action for TS 3.1.4, 3.1.5 and 3.1.6.
Choice D is correct, see explanation above

Question 84 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11144
User-Defined ID:	SE10015-N01
Cross Reference Number:	BWLI-RC1-143
Topic:	SE10015-N01
Num Field 1:	3.4
Num Field 2:	4.7
Text Field:	E/APE015 2.2.40
Comments:	Braidwood NRC Exam 2009 Question # 84 015/017 Reactor Coolant Pump (RCP) Malfunctions Ability to apply Technical Specifications for a system. TIER: 1 GROUP: 1 Cog Level: High Task No: S-TS-007 Obj No: 8E.TS-007 Cross Ref: 10CFR55.43(b) (2)

General Data	
Technical Reference with Revision Number:	TS LCO 3.4.5
Justification for Non SRO CFR Link:	
Question History:	From Beaver Valley previous NRC exam bank question (Focus Learning ILT exam bank ID # 29467) also similar to Braidwood ILT bank # BWLI-RC1-143. Not used in current ILT class.
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	Added RCS dilution to stem to add plausibility to distractor. Reworded answer and distractors to match Bwd LCO action statements.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

85 **ID: SE10038-N01** **Points: 1.00**

Operating within Tech Spec limiting conditions, ensures that during a steam generator tube rupture the dose limits of 10 CFR 50.67 are NOT exceeded. These limits are ____(1)___ rem TEDE for an individual at the boundary of the exclusion area for 2 hours OR ____(2)___ rem TEDE for an individual in the Main Control Room for the duration of the event.

A. (1) 50
(2) 25

B. (1) 10
(2) 15

C. (1) 15
(2) 10

D. (1) 25
(2) 5

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of safety limits found in the bases for LCO 3.4.16 RCS specific activity.
The question is SRO level because requires knowledge of the Tech Spec bases of LCO 3.4.16.
10 CFR 50.67 lists 25 rem TEDE to any individual at the boundary of the exclusion area and 5 rem TEDE for an occupancy of the MCR.
Distractor of 10 rem TEDE is the dose limit for protecting private property.
Distractor of 15 REM is LDE Federal limit (Lens Dose Equivalent limit per year).
Distractor of 50 is TODE Federal limit.
Choice A is incorrect, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is correct, see explanation above.

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:	11145
User-Defined ID:	SE10038-N01
Cross Reference Number:	
Topic:	SE10038-N01
Num Field 1:	4.0
Num Field 2:	4.7
Text Field:	E/APE038 2.2.22
Comments:	Braidwood NRC Exam 2009 Question # 85 038 Steam Generator Tube Rupture (SGTR) Knowledge of limiting conditions for operations and safety limits. TIER: 1 GROUP: 1 Cog Level: Low Task No: S-TS-006 Obj No: 8E.TS-006 Cross Ref: 10CFR55.43(b) (2)

General Data	
Technical Reference with Revision Number:	LCO 3.4.16 bases 10 CFR 50.67 ILT lesson plan I1-BZ-XL-01 rev. 2
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

86 **ID: SE10040-N01** **Points: 1.00**

Given:
- Unit 1 is at 100% power normal alignment.

An event occurs and one minute later the following indications are noted:
- Annunciator 1-1-A2, CNMT DRAIN LEAK DETECT FLOW HIGH is in alarm.
- Reactor Power is 102% and slowly rising.
- Auctioneered Tavg is 585°F and slowly lowering.
- Containment Pressure is approximately 2 psig and slowly rising.
- Pressurizer level is 57% and slowly lowering.
- Pressurizer pressure is 2210 psig and slowly lowering.

With the above indications, the US will direct the crew to...

A. trip the reactor and enter 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, because there is a primary RCS break in containment.

B. enter 1BwOA PRI-16, RESPONSE TO OVERPOWER CONDITION, and reduce turbine load because there is a dilution event in progress.

C. enter 1BwOA PRI-1, EXCESSIVE PRIMARY PLANT LEAKAGE, because there is a primary RCS break in containment.

D. trip the reactor and enter 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, because there is a secondary steam break in containment.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee ability to interpret steam line rupture conditions requiring a reactor trip.
The question is SRO level because it requires assessment of conditions and selecting of procedures.
Choice A is incorrect, a RCS break would not result in rising reactor power or lowering RCS temperature.
Choice B is incorrect, although 1BwOA PRI-16 entry conditions are met, the event is not a dilution because dilutions would raise RCS temperature instead of lowering it. Also dilution would not cause severe Prz pressure and level changes.
Choice C is incorrect, a 3% Prz level drop in one minute would easily exceed the capacity of a charging pump, therefore even if this was an RCS leak, PRI-1 would not be appropriate.
Choice D is correct, steam break will account for all indications given.

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:	11146
User-Defined ID:	SE10040-N01
Cross Reference Number:	
Topic:	SE10040-N01
Num Field 1:	4.6
Num Field 2:	4.7
Text Field:	E/APE040 A2.02
Comments:	Braidwood NRC Exam 2009 Question # 86 040 Steam Line Rupture Ability to determine and interpret the following as they apply to the Steam Line Rupture: Conditions requiring a reactor trip TIER: 1 GROUP: 1 Cog Level: High Task No: S-EP-001 Obj No: 7D.EP-002-D Cross Ref: 10CFR55.43(b)(5)

General Data	
Technical Reference with Revision Number:	ILT lesson plan I1-MS-XL-01 rev. 4a
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

87 **ID: SE10054-N01** **Points: 1.00**

Given:
- Unit 1 was operating at 100% power, normal alignment for the past 6 months.
- 30 minutes ago, a feedwater malfunction caused the crew to perform a CD/FW RUNBACK from 100% power.

With the above condition, which of the following LCO/TRM surveillance frequencies specifically require an additional performance of the surveillance because of the unit ramp?
(assume all required pre-transient surveillances were completed within the 12 hours preceding the ramp)

A. 3.4.b RCS CHEMISTRY verifying RCS chemistry is within limits.

B. 3.4.13 RCS OPERATIONAL LEAKAGE verifying operational leakage is within limits.

C. 3.4.16 RCS SPECIFIC ACTIVITY verifying dose equivalent I-131 within limits.

D. 3.7.3 SECONDARY SPECIFIC ACTIVITY verifying dose equivalent I-131 within limits.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee ability to apply Tech Specs for a system due to a loss of feedwater.
The question is SRO level because it is the SROs responsibility to ensure the surv. frequency is met by contacting the chemistry department following a turbine ramp of $\geq$ 15% RTP within a 1 hour period.

Choice A is incorrect, RCS chemistry surv frequency for chlorides, fluoride and dissolved oxygen are in accordance with EPRI PWR Primary Water Chemistry Guidelines. These frequencies are designated in chemistry procedure CY-AP-120-100 with no specific requirement for additional sampling because of a unit ramp. However, the SRO does not need to know the chemistry procedure, the knowledge requirement is only that there is no frequency requirement change in the TRM because of a unit ramp.
Choice B is incorrect, RCS operational leakage frequency is every 72 hours. There is a note that says the surv. is not performed until 12 hours of steady state ops, however that does not increase the frequency, only states that 12 hours of steady state ops must be complete prior to the surv. performance.
Choice C is correct, RCS activity surv. requirement for I-131 is 14 days AND within 2 to 6 hours after a ramp of $\geq$ 15% RTP within a 1 hour period. The CD/FW runback would ramp the unit from 100% to approx. 700 MW at 250 MW/min. This would exceed the rate of $\geq$ 15% RTP within a 1 hour period, thus requiring the change in frequency to within 2 to 6 hrs following the ramp.
Choice D is incorrect, secondary specific activity surv. for I-131 has a frequency of every 31 days with no specific frequency requirement change because of a unit ramp.

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:	11147
User-Defined ID:	SE10054-N01
Cross Reference Number:	
Topic:	SE10054-N01
Num Field 1:	3.4
Num Field 2:	4.7
Text Field:	E/APE054 2.2.40
Comments:	Braidwood NRC Exam 2009 Question # 87 054 Loss of Main Feedwater (MFW) Ability to apply Technical Specifications for a system. TIER: 1 GROUP: 1 Cog Level: High Task No: S-TS-008 Obj No: 8E.TS-008 Cross Ref: 10CFR55.43(b) (2)

General Data	
Technical Reference with Revision Number:	Tech Spec 3.4.16
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

88 **ID: SE10009-N01** **Points: 1.00**

Given:
- Unit 1 has experienced a reactor trip and SI from 100% power.
- All equipment operated as designed.
- The crew has transitioned to 1BwEP ES-1.1, "SI TERMINATION" from 1BwEP-1 "LOSS OF REACTOR OR SECONDARY COOLANT".
- At step 4 of 1BwEP ES-1.1, the team secures all but one charging pump as directed.
- After one charging pump is secured, RCS Pressure begins to slowly and steadily LOWER.

With the above conditions, the US will direct the crew to...

A. manually re-initiate SI and transition to 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION.

B. re-start the charging pump to stabilize RCS pressure and continue with 1BwEP ES-1.1.

C. transition to 1BwEP ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

D. re-start the charging pump to stabilize RCS pressure and transition back to 1BwEP-1.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee ability to interpret RCS parameters as they apply to a small break LOCA.
The question is SRO level because it requires assessment of conditions and selecting of procedures.
In 1BwEP ES-1.1, step 5 checks RCS pressure following the securing of one charging pump. If pressure is lowering, the RNO column transitions crew to 1BwEP ES1.2.

Choice A is incorrect, this would be correct if subcooling or Prz level could not be maintained in EP ES-1.1.
Choice B is incorrect, restarting pumps and continuing with the current procedure would be correct if subcooling or Prz level could not be maintained in EP ES-1.2.
Choice C is correct, see explanation above
Choice D is incorrect, this would be correct if subcooling or Prz level could not be maintained following the shut down of the RH pumps in step 10, per the OAS of EP ES-1.1, but not at step 5.

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:	11148
User-Defined ID:	SE10009-N01
Cross Reference Number:	
Topic:	SE10009-N01
Num Field 1:	3.3
Num Field 2:	3.4
Text Field:	E/APE009 A2.15
Comments:	Braidwood NRC Exam 2009 Question # 88 009 Small Break LOCA Ability to determine or interpret the following as they apply to a small break LOCA: RCS parameters TIER: 1 GROUP: 1 Cog Level: High Task No: S-EP-040 Obj No: 7D.EP-002-F Cross Ref: 10CFR55.43(b)(5)

General Data	
Technical Reference with Revision Number:	1BwEP ES-1.1 rev. 201
Justification for Non SRO CFR Link:	
Question History:	From Surry previous NRC exam bank question (Focus Learning ILT exam bank ID # 25786)
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

89 **ID: SE10062-N01** **Points: 1.00**

Given the following initial conditions:
- Unit 1 at 100% power.
- 1B CV pump danger tagged OOS, 24 hours ago and LCO 3.5.2, ECCS OPERATING has been entered.
- 0B Fire Pump is danger tagged OOS.
- All other systems are normally aligned.

The following event occurs:

- The SX header supply to "1A" train ECCS pumps ruptures and manual valve 1SX013A, SX SUP TO TRAIN A HXs, is locally closed, isolating SX to ALL the 1A train ECCS pumps AND cubicle coolers.
- The crew entered LCO 3.7.8, SX SYSTEM and is performing 1BwOA PRI-8, ESSENTIAL SERVICE WATER MALFUNCTION.
- An EO is having difficulty aligning fire protection (FP) to the 1A CV pump oil coolers.

- At 0901 hours, the 1A CV pump temperatures are:
  All pump bearing temps - 130°F and RISING at 5°F/min.
  Gear drive oil temp - 95°F and RISING at 10°F/min.

(assume the rates of temperature rise remain constant unless cooling is restored)

(1) Assuming FP is NOT aligned to the 1A CV pump oil coolers, 1A CV pump temperatures will initially reach trip limits at...
Note: 1BwOA PRI-8, Table A is attached.

(2) Assuming FP IS aligned to the 1A CV pump oil coolers prior to reaching a pump trip limit AND the plant remains in that configuration, per Tech Specs the 1A CV pump will be considered...

A. (1) 0909
(2) OPERABLE with no loss of safety function.

B. (1) 0909
(2) INOPERABLE because a loss of safety function exists.

C. (1) 0916
(2) OPERABLE with no loss of safety function.

D. (1) 0916
(2) INOPERABLE because a loss of safety function exists.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to determine length of time after loss of SX to CV pump that the pump may be damaged (require trip).
The question is SRO level because it requires the examinee to make a operability determination of the CV pump which is an SRO function.
For part (1), when calculating time to trip criteria for pump temps formula is (trip temp - current temp)/rate of rise.
For bearing temp: (205°F - 130°F) / 5°F/min = 15 min.
For gear drive oil temp: (175°F - 95°F) / 10°F/min = 8 min.

Shortest time is 0901 + 8 min. = 0909
For part (2), per the safety function determination program, SX is a support system of ECCS (CV pump). A supported SSC (CV pump) is inoperable when the support system (SX) is not capable of performing its intended function. Although FP would supply cooling to the CV pumps during normal operations, it is a non-safety related system that can not be relied upon during a DBA, therefore the 1A CV pump is considered inoperable.

Choice A is incorrect, 1A CV pump would be considered inoperable due to loss of safety function.
Choice B is correct, see explanation above.
Choice C is incorrect, wrong time in part (1), and wrong operability determination in part (2).
Choice D is incorrect, wrong time in part (1).

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:	11149
User-Defined ID:	SE10062-N01
Cross Reference Number:	
Topic:	SE10062-N01
Num Field 1:	3.6
Num Field 2:	4.2
Text Field:	E/APE062 A2.06
Comments:	Braidwood NRC Exam 2009 Question # 89 062 Loss of Nuclear Service Water Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is de-creasing TIER: 1 GROUP: 1 Cog Level: High Task No: S-TS-007 Obj No: 7E.TS-010-D Cross Ref: 10CFR55.43(b)(2)

General Data	
Technical Reference with Revision Number:	1BwOA PRI-8, Table A rev. 102 TRM appendix "O" SFDP
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	1BwOA PRI-8, Table A
Excluded Reference:	

90 **ID: SE10003-N01** **Points: 1.00**

Given:
- Unit 1 was at 100% power normal alignment.

An event occurs and the following indications are noted at 1PM05J.
- Multiple block 10 annunciators are in alarm.
- Auctioneered high Tave is 4°F BELOW Tref and stable.
- ONE DRPI Rod Bottom light is LIT.
- Control rods bank D automatically stepped OUT 3 steps and then STOPPED stepping.
- Power Range NIs are stable at:

N41- 101%
N42- 101%
N43- 77%
N44- 105%

With the above conditions the US will direct the crew to...

A. enter 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, due to the NI channel indications.

B. enter 1BwOA ROD-1, UNCONTROLLED ROD MOTION, due to the outward rod motion.

C. enter 1BwOA ROD-2, FAILURE OF RODS TO MOVE, due to the Tave/Tref mismatch.

D. enter 1BwOA ROD-3, DROPPED OR MISALIGNED ROD, due to the DRPI indications.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee ability to interpret a dropped rod using ex-core instruments and loop temperature measurements.
The question is SRO level because it requires assessment of conditions and selection of appropriate procedure.
The given indications are all consistent with a dropped control rod.
Choice A is incorrect, No power range NI channels have exceeded the reactor trip setpoint of 108%.
Choice B is incorrect, outward rod motion would be expected due to the Tave/Tref mismatch created from the dropped rod.
Choice C is incorrect, rods are failing to move because of rod stop and C-11.
Choice D is correct, All indications are consistent with a single dropped rod.

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:	11150
User-Defined ID:	SE10003-N01
Cross Reference Number:	
Topic:	SE10003-N01
Num Field 1:	3.6
Num Field 2:	3.8
Text Field:	E/APE003 A2.03
Comments:	Braidwood NRC Exam 2009 Question # 90 003 Dropped Control Rod Ability to determine and interpret the following as they apply to the Dropped Control Rod: Dropped rod, using in-core/ex-core instrumentation, in-core or loop temperature measurements TIER: 1 GROUP: 2 Cog Level: High Task No: S-OA-091 Obj No: 7D.OA-033-A Cross Ref: 10CFR55.43(b) (5)

General Data	
Technical Reference with Revision Number:	ILT lesson plan I1-OA-XL-34 rev. 9a
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

91 **ID: SE1WE01-N01** **Points: 1.00**

Given:
- A reactor trip and Safety Injection have occurred on Unit 1.
- The crew was performing actions of 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.
- Due to conflicting indications, the crew entered 1BwEP ES-0.0, REDIAGNOSIS.
- While performing 1BwEP ES-0.0, the crew determines that there is a rising trend on secondary plant radiation monitors AND one SG level is rising faster than the other three with Auxiliary Feedwater flow being equal.

Which one of the following describes how the US will transition from 1BwEP ES-0.0, REDIAGNOSIS to the correct procedure?

A. Return to 1BwEP-0 diagnostic steps to verify indications then transition to 1BwEP-3.

B. Go directly to 1BwEP-3 or 1BwCA-3 series procedure.

C. Return to 1BwEP-1 step in effect and use the Operator Action Summary page for transition to 1BwEP-3.

D. Use Heat Sink status tree for transition to 1BwFR-H.3, RESPONSE TO STEAM GENERATOR HIGH LEVEL.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to select appropriate procedure transition during performance of Rediagnosis.
The question is SRO level because it requires selection of the proper procedure flowpath.
Choice A is incorrect, there is no transition back to 1BwEP-0 from 1BwEP ES-0.0.
Choice B is correct, 1BwEP ES-0.0, step 4 directs transition to 1BwEP-3 or 1BwCA-3 series procedures.
Choice C is incorrect, transition back to 1BwEP-1 from 1BwEP ES-0.0 is required if NO indication of a SG tube leak are noted.
Choice D is incorrect, 1BwFR H-3 is a yellow path for high SG level but yellow paths do not require immediate operator action. In this case the rising SG level will be mitigated by strategy of 1BwEP-3.

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:	11151
User-Defined ID:	SE1WE01-N01
Cross Reference Number:	
Topic:	SE1WE01-N01
Num Field 1:	3.2
Num Field 2:	4.0
Text Field:	W/E01 A2.01
Comments:	Braidwood NRC Exam 2009 Question # 91 E01 Rediagnosis Ability to determine and interpret the following as they apply to the (Reactor Trip or Safety Injection Rediagnosis): Facility conditions and selection of appropriate procedures during abnormal and emergency operations. TIER: 1 GROUP: 2 Cog Level: Low Task No: S-EP-006 Obj No: 7D.EP-001-B Cross Ref: 10CFR55.43(b)(5)

General Data	
Technical Reference with Revision Number:	1BwEP ES-0.0 rev. 200
Justification for Non SRO CFR Link:	
Question History:	From Beaver Valley previous NRC exam bank question (Focus Learning ILT exam bank ID # 29564)
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

92 **ID: SE10074-N01** **Points: 1.00**

Given:
- An inadequate core cooling event is in progress on Unit 1.
- The crew is performing 1BwFR-C.1 RESPONSE TO INADEQUATE CORE COOLING.
- At step 3, CHECK RCP SUPPORT CONDITIONS, support conditions were NOT established due to no seal injection flow and high RCP leakoff temperatures.
- Continued attempts to re-establish RCP support conditions have been UNSUCCESSFUL so far.
- Currently, the crew is at step 16, CHECK IF RCPs SHOULD BE STARTED.
- All RCPs are shutdown and the RCS loops are available.
- CETCs are 1235°F and slowly rising.

With the above conditions, the US will direct the crew to...

A. immediately start an RCP regardless of the support conditions.

B. remain at step 16 until support conditions are established, then start an RCP.

C. NOT start an RCP and continue with the procedure using another mitigation strategy.

D. continue with the procedure while attempting to establish support conditions, then return to step 16 when support conditions are established.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of operational implication of EOP note during inadequate core cooling.
The question is SRO level because it requires assessment of conditions and direct appropriate MCR actions.
Note prior to step 16 reads "Normal conditions are desired but NOT required for starting RCPs". Operational implication is to start an RCP regardless of support conditions.

Choice A is correct, RCP is started to provide temporary cooling to avoid core damage while some form of RCS injection flow is established.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

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:	11152
User-Defined ID:	SE10074-N01
Cross Reference Number:	
Topic:	SE10074-N01
Num Field 1:	3.8
Num Field 2:	4.3
Text Field:	E/APE074 2.4.20
Comments:	Braidwood NRC Exam 2009 Question # 92 074 Inadequate Core Cooling Knowledge of the operational implications of EOP warnings, cautions, and notes. TIER: 1 GROUP: 2 Cog Level: High Task No: S-FR-009 Obj No: 7D.FR-002-A Cross Ref: 10CFR55.41(b)(10)

General Data	
Technical Reference with Revision Number:	1BwFR-C.1 rev. 200
Justification for Non SRO CFR Link:	The question is SRO level because in the accident conditions described the SRO would be directing the crew operations with a procedure step so critical to reactor safety, especially when the action would in all likeliness "sacrifice" the RCP.
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

93 **ID: SE1WE10-N01** **Points: 1.00**

Given:
- Unit 1 experienced a faulted SG in containment.
- The crew is performing 1BwEP ES-0.2, NATURAL CIRCULATION COOLDOWN.

The STA monitoring the Critical Safety Functions notes the following indications:

- Reactor Power	6 %
- SUR on both IR and SR channels	0.2 DPM
- Containment Pressure	21 psig
- CETCs Average of 10 highest	280°F
- RCS Pressure	480 psig

	1A	1B	1C	1D
- RCS Cold leg temperatures.	245°F	260°F	275°F	275°F
- Narrow Range S/G Levels	0%	40%	42%	41%
- Total available Feed Flow to S/Gs	0 gpm	0 gpm	0 gpm	0 gpm

With the above indications, the US will transition to...

A. 1BwFR S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS.

B. 1BwFR H.1, RESPONSE TO LOSS OF SECONDARY HEAT SINK.

C. 1BwFR P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION.

D. 1BwFR Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of parameters and logic used to assess status of safety functions.
The question is SRO level because it requires assessment of plant conditions and selection of emergency procedure.

Choice A is correct, Highest safety function red path is subcriticality. Power >5%.
Choice B is incorrect, Red path for heat sink is all SG levels below 31% (adverse containment) with < 500 gpm AF flow..
Choice C is incorrect, Red path for integrity is all RCS cold leg temperatures < limit A curve (which is less than the 240 deg. line.
Choice D is incorrect, The highest containment safety function path can be is orange at 21 psig.

Question 93 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11153
User-Defined ID:	SE1WE10-N01
Cross Reference Number:	BWLI-ST-030
Topic:	SE1WE10-N01
Num Field 1:	4.0
Num Field 2:	4.6
Text Field:	W/E10 2.4.21

Comments:

Braidwood NRC Exam 2009 Question # 93
E10 Natural Circulation with Steam Void in Vessel with/without
RVLIS
Knowledge of the parameters and logic used to assess the status of safety |
functions, such as reactivity control, core cooling and heat removal, reactor |
coolant system integrity, containment conditions, radioactivity release control, |
etc. |

TIER:	1
GROUP:	2
Cog Level:	High
Task No:	S-ST-003
Obj No:	8D.ST-003
Cross Ref:	10CFR55.43(b)(5)

General Data	
Technical Reference with Revision Number:	1BwST-1, 3, 4, 5 revs. 200
Justification for Non SRO CFR Link:	
Question History:	Bwd ILT Bank #BWLI-ST-030, not used in current ILT class.
Question Source:	Modified
Low KA Justification (if required):	
Revision History:	changed initial conditions from EP-1 to EP ES-0.2 modified conditions to change answer from 1BwFR H.1 to 1BwFR S.1
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

94 **ID: SG10020-N01** **Points: 1.00**

Per BwAP 340-1, USE OF PROCEDURES FOR OPERATING DEPARTMENT, which of the following correctly describe when an emergency procedure step on the Continuous Action Summary page, is applicable?

A. Only PRIOR to performing the step in the main body of the procedure, AND it MAY apply after a transition is made to another procedure.

B. Only after proceeding PAST the step in the main body of the procedure, AND it MAY apply after a transition is made to another procedure.

C. Only after proceeding PAST the step in the main body of the procedure, BUT it will NEVER apply after a transition is made to another procedure.

D. ANY time during the procedure performance, BUT it will NEVER apply after a transition is made to another procedure.

Answer: B

Answer Explanation:

The question meets the K/A, requires examinee ability to interpret and execute procedure steps. Examinee must know rules of usage for continuous action steps in order to properly execute them.
The question is SRO level because BwAP 340-1, section C.4.b.4) specifically lists monitoring status of continuous action summary page steps as an SRO responsibility. Additionally, BwAP 340-1 states that a step on the CAS page becomes applicable AFTER proceeding past that step in the main body. A CAS step then remains applicable until it is superseded by alternate guidance or stated to be inapplicable. It generally remains applicable throughout its associated procedure unless otherwise stated, and may apply after transition to another procedure if the actions are not inappropriate for the other procedure.

Choice A is incorrect, see explanation above also it is plausible because the OAS (vs. CAS) page list procedure steps that are applicable prior to performing it in the procedure main body.
Choice B is correct, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

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:	11154
User-Defined ID:	SG10020-N01
Cross Reference Number:	
Topic:	SG10020-N01
Num Field 1:	4.6
Num Field 2:	4.6
Text Field:	GEN 2.1.20
Comments:	Braidwood NRC Exam 2009 Question # 94 Ability to interpret and execute procedure steps. TIER: 3 Category: 1 Cog Level: low Task No: S-AM-003 Obj No: 7E.AM-003-A Cross Ref: 10CFR55.41(b)(10)

General Data	
Technical Reference with Revision Number:	BwAP 340-1 rev. 23
Justification for Non SRO CFR Link:	BwAP 340-1 specifically list question topic as SRO responsibility.
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

95 **ID: SG10009-N01** **Points: 1.00**

Given:
- The Unit 1 Unit Supervisor determines that the number of people in the control room is creating a distraction.

Per OP-AA-103-101, CONTROL ROOM ACCESS CONTROL, which of the following personnel will the US direct to leave the control room?

A. A Nuclear Oversight auditor who is performing Control Room observations.

B. The Station Manager who is performing "day-in-the-plant" observations.

C. The Illinois Dept. of Nuclear Safety inspector and the admin NSO who are reviewing an IDNS report.

D. The Field Supervisor and an Equipment Operator who are performing a pre-job brief of an oil separator surveillance.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee ability to direct control room activities by knowing the restrictions to control room access.

Choice A is incorrect, NOS personnel are specifically listed as having unlimited access to the MCR in OP-AA-103-101.
Choice B is incorrect, the Station Manager is specifically listed as having unlimited access to the MCR in OP-AA-103-101.
Choice C is incorrect, IDNS personnel are "regulatory personnel" who are specifically listed as having unlimited access to the MCR in OP-AA-103-101.
Choice D is correct, While on-shift operations personnel are normal provided unlimited access to the control room, the access is limited to personnel conducting control room business (unless assign to a position in the MCR). In this case the FS and EO should brief the surveillance elsewhere since an oil separator surveillance has no interface with the MCR.

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:	11155
User-Defined ID:	SG10009-N01
Cross Reference Number:	BWLI-AA01003
Topic:	SG10009-N01
Num Field 1:	2.9
Num Field 2:	4.5
Text Field:	GEN 2.1.09
Comments:	Braidwood NRC Exam 2009 Question # 95 Ability to direct personnel activities inside the control room. TIER: 3 Category: 1 Cog Level: Low Task No: S-AM-206 Obj No: 8E.AM-217 Cross Ref: 10CFR55.41(b) (10)

General Data	
Technical Reference with Revision Number:	OP-AA-103-101 rev. 1
Justification for Non SRO CFR Link:	The question is SRO level because OP-AA-103-101 section 3.1 states "Access to the Control Room is the ultimate responsibility of, and shall be controlled by the Unit Supervisor."
Question History:	Previous version used in Bwd ILT NOPS phase
Question Source:	Modified from Bwd ILT bank #BWLI-AA01003
Low KA Justification (if required):	
Revision History:	Revised stem to make SRO level. Revised answer from security supervisor to operations personnel that are conducting business that is not related to the control room.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

96 **ID: SG20011-N01** **Points: 1.00**

Given:
- An SRO is screening the following shift activities for possible temporary configuration changes.

Per CC-AA-112, TEMPORARY CONFIGURATION CHANGES, which activity will the US have to enter into the Temp Change Log?

A. A hose is being connected to the RCS to support draining activities per the operating procedure.

B. Thermal overloads are being removed from a breaker that are specified on a clearance order.

C. An electrical jumper is being placed in a component's start circuit as part of a surveillance procedure to prevent inadvertent auto start.

D. A blocking device is being installed on a safety valve to stop leakage for which a work request has been written.

Answer: D

Answer Explanation:

The question meets the K/A, requires examinee knowledge of process for controlling temporary design changes.
The question is SRO level because to screen configuration changes and maintain the temp change log.

Choice A is incorrect, CC-AA-112 section 4.1.2.9 list drain hoses connected as part of an approved procedure, as an exclusion from the TCC process.
Choice B is incorrect, CC-AA-112 section 4.1.2.5 list configuration changes to SSCs included within an operations clearance as an exclusion from the TCC process.
Choice C is incorrect, CC-AA-112 section 4.1.2.2 list jumper to conduct a test that is part of a approved surveillance, as an exclusion from the TCC process.
Choice D is correct, CC-AA-112 section 4.1.3 list gagging or disabling relief valves as NOT an exclusion from the TCC process.

Question 96 Info	
Question Type:	Multiple Choice
Status:	Active
Always select on test?	No
Authorized for practice?	No
Points:	1.00
Time to Complete:	1
Difficulty:	0.00
System ID:	11156
User-Defined ID:	SG20011-N01
Cross Reference Number:	BWLI-AA01118
Topic:	SG20011-N01
Num Field 1:	2.3
Num Field 2:	3.3
Text Field:	GEN 2.2.11
Comments:	Braidwood NRC Exam 2009 Question # 96 Knowledge of the process for controlling temporary design changes. TIER: 3 Category: 2 Cog Level: Low Task No: S-AM-186 Obj No: 8E.AM-186 Cross Ref: 10CFR55.43(b)(3)

General Data	
Technical Reference with Revision Number:	CC-AA-112 rev. 14
Justification for Non SRO CFR Link:	
Question History:	Bwd ILT bank #BWLI-AA01118, not currently used in the ILT program.
Question Source:	Bank
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

97 **ID: SG20019-N01** **Points: 1.00**

Given:
- Work planning has contacted an SRO to discuss the need to modify the scope of a work order, that is attached to an active clearance order (C/O).
- Based on this discussion, the SRO determines that the C/O will need to be revised before the new work can take place.

With the above conditions, prior to revising, the SRO will ensure the electronic C/O has a Holder Acceptance Lock applied OR the C/O status is placed in...

A. SUSPEND.

B. APPROVED.

C. TEMP LIFT.

D. DISTRIBUTED.

Answer: A

Answer Explanation:

The question meets the K/A, requires examinee knowledge of maintenance work order requirements.
The question is SRO level because OP-AA-109-101 section 9.3.1 lists the task as an Operations Supervisor responsibility.

Choice A is correct, OP-AA-109-101 section 9.3.1 directs the SRO to ensure the status is SUSPEND. This status removes clearance order coverage and no work requiring C/O protection can proceed.
Choice B is incorrect, APPROVED is final approval granted.
Choice C is incorrect, TEMP LIFT is C/O process used to temporarily restore equipment to service for testing.
Choice D is incorrect, DISTRIBUTED is checklist ready to be executed.

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:	11157
User-Defined ID:	SG20019-N01
Cross Reference Number:	
Topic:	SG20019-N01
Num Field 1:	2.3
Num Field 2:	3.4
Text Field:	GEN 2.2.19
Comments:	Braidwood NRC Exam 2009 Question # 97 Knowledge of maintenance work order requirements. TIER: 3 Category: 2 Cog Level: Low Task No: S-AM-033 Obj No: 7E.AM-033-A Cross Ref: 10CFR55.43(b)(3)

General Data	
Technical Reference with Revision Number:	OP-AA-109-101 rev. rev. 4
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

98 **ID: SG30005-N01** **Points: 1.00**

Given the following plant conditions on **Unit 2:**

- A reactor trip and SI occurred 6 hours ago due to a small break RCS LOCA.
- CETCs are 675 °F.
- BOTH SR channels are 1500 cps and lowering.
- RCS pressure is 600 psig.
- ALL RCPs are tripped.
- RVLIS plenum level is 15%.
- Cnmt floor water level is 20 inches.
- ALL S/G NR levels are 35%.
- Cnmt rad levels (2AR020/021) are 1500 R/hr.
- Cnmt pressure is 13 psig.
- NO offsite release has occurred, NOR is one anticipated.

Currently, with the above conditions, this event will be classified as a(n) ...

(EP-AA-1001 Braidwood Annex is provided for reference)

A. Unusual Event.

B. Alert.

C. Site Area Emergency.

D. General Emergency.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee ability to use radiation monitoring system (to assess EAL).
The question is SRO level because EAL calls are an SRO function.
Given information indicates loss of RCS (small break LOCA) and loss of Fuel Clad (containment rads exceed threshold of table F1 for fuel clad). Loss of any two fission product barriers is FS1 (site area emergency)

Choice A is incorrect, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is correct, see explanation above.
Choice D is incorrect, containment is NOT a potential loss. Rad levels do NOT exceed threshold in table F1 for containment.

Question 98 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:	11158
User-Defined ID:	SG30005-N01
Cross Reference Number:	BWLC7FZP1013
Topic:	SG30005-N01
Num Field 1:	2.9
Num Field 2:	2.9
Text Field:	GEN 2.3.05
Comments:	Braidwood NRC Exam 2009 Question # 98 Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. TIER: 3 Category: 3 Cog Level: High Task No: S-ZP-008 Obj No: 8F.ZP-008 Cross Ref: 10CFR55.43(b)(5)

General Data	
Technical Reference with Revision Number:	EP-AA-1001 rev. 22
Justification for Non SRO CFR Link:	
Question History:	none
Question Source:	Modified from Bwd LORT bank #BWLC7FZP1013
Low KA Justification (if required):	
Revision History:	Modified to change answer from alert to site area emergency. Changed time after event from 35 min. to 6 hrs. Changed containment rads from 510 R/hr to 1500 R/hr. Added stem line that no release occurring.
ILT	
Supplied Ref (If appropriate):	EP-AA-1001 rev. 22
Excluded Reference:	

99 **ID: SG30013-N01** **Points: 1.00**

Given:
- Unit 1 is at 99% power, normally aligned.
- The on-coming Unit 1 SRO, reviewing the turnover, notes several activities pending during the next shift.
- Shortly after shift turnover, Rad Protection requests that a containment entry be made during the shift, to obtain rad conditions, for upcoming maintenance on a pressurizer spray valve.

In accordance with BwAP 1450-1, ACCESS TO CONTAINMENT, which one of the following pending Unit 1 activities, must the US ensure does NOT occur during the containment entry?

A. A flux map.

B. A containment release.

C. Raise turbine load by 5 MW.

D. A 10 gallon boration.

Answer: A

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 BwAP 1450-1 attachment 2, Containment Entry Checklist, specifically requires operations supervisor to verify the incore detectors are OOS and approve the containment entry.

Choice A is correct, a flux map would require the incore detectors be returned to service and energized, which is prohibited by BwAP 1450-1 section F.3.c (unless maintenance is being performed on the MIDS.
Choice B is incorrect, a containment release does not have any restrictions during containment entry.
Choice C is incorrect, BwAP 1450-1 requires maintaining stable reactor power during the containment entry, however, stable reactor power is defined as $\pm$ 2% RTP. Raising turbine power by 10 MW would be less than 1% power rise.
Choice D is incorrect, BwAP 1450-1 section F.3.b.2 allows boric acid adjustments while maintaining reactor power within stable limits.

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:	11159
User-Defined ID:	SG30013-N01
Cross Reference Number:	
Topic:	SG30013-N01
Num Field 1:	3.4
Num Field 2:	3.8
Text Field:	GEN 2.3.13
Comments:	Braidwood NRC Exam 2009 Question # 99 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. TIER: 3 Category: 3 Cog Level: High Task No: S-AM-128 Obj No: 7E.AM-128-A Cross Ref: 10CFR55.43(b)(4)

General Data	
Technical Reference with Revision Number:	BwAP 1450-1 rev. 35
Justification for Non SRO CFR Link:	
Question History:	Prairie Island previous NRC exam
Question Source:	Modified from Prairie Island previous NRC exam bank question (Focus Learning ILT exam bank ID # 26140)
Low KA Justification (if required):	
Revision History:	Changed all three distractors from containment rad levels and power level.
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	

100 **ID: SG40011-N01** **Points: 1.00**

Given:
- Unit 1 is in mode 4 during a refuel outage.
- 1A RH train is in shutdown cooling mode.
- 1B RH Pump is OOS.

The following event occurs:

- The I/P converter for 1RH606, HX 1A FLOW CONT VLV fails such that the valve goes full open.

- The Unit 1 SRO has dispatched an equipment operator (EO) to install a temporary regulator, with a pneumatic jumper, around the failed I/P converter to re-enable RH temperature control.

With the above conditions, the US will direct the EO to follow the pneumatic jumper installation instructions found in...

A. 1BwOA SEC-4, LOSS OF INSTRUMENT AIR.

B. 1BwOA ELEC-2, LOSS OF INSTRUMENT BUS.

C. 1BwOA PRI-5, CONTROL ROOM INACCESSIBILITY.

D. 1BwOA PRI-10, LOSS OF RH COOLING.

Answer: C

Answer Explanation:

The question meets the K/A, requires examinee knowledge of abnormal condition procedures.
The question is SRO level because it requires assessment of conditions and selection of appropriate procedure.
The installation instructions for pneumatic jumpers for RH system are found in 1BwOA PRI-5, Attachment C. This operation is in PRI-5 to allow cooldown of the plant to mode 5 while operating from the remote shutdown panel because there are no RH controls in the remote shutdown panel.
Choice A is incorrect, SEC-4 is a credible distractor because a loss of instrument air would also affect the RH flow control valves, however, the mitigation strategy is to secure the RH train and use alternate modes of cooling per PRI-10.
Choice B is incorrect, ELEC-2 is a credible distractor because it contains steps for loss of control power to the RH Hx flow control valves.
Choice C is correct, see explanation above.
Choice D is incorrect, PRI-10 is a credible distractor because it contains mitigation strategies when RH cooling is lost, however in this question, RH cooling would be maximized instead of lost.

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:	11160
User-Defined ID:	SG40011-N01
Cross Reference Number:	
Topic:	SG40011-N01
Num Field 1:	4.0
Num Field 2:	4.2
Text Field:	GEN 2.4.11
Comments:	Braidwood NRC Exam 2009 Question # 100 Knowledge of abnormal condition procedures. TIER: 3 Category: 4 Cog Level: Low Task No: S-OA-040 Obj No: 8D.OA-015-B Cross Ref: 10CFR55.43(b) (5)

General Data	
Technical Reference with Revision Number:	1BwOA PRI-5 rev. 103
Justification for Non SRO CFR Link:	
Question History:	None
Question Source:	New
Low KA Justification (if required):	
Revision History:	
ILT	
Supplied Ref (If appropriate):	None
Excluded Reference:	