EXAMINATION ANSWER KEY

2016 NRC

1	ID: 1267496	Points: 1.00

Braidwood U-1 is being ramped from 50% to 75% reactor power.
During this ramp, RCS loop ΔTs will change from their initial value (in degrees F.) of ____________

A. 15°F to 22.5°F.

B. 30°F to 45°F.

C. 50°F to 75°F.

D. 66°F to 99°F.

Answer: B

Answer Explanation
Bwd 2016 NRC Exam Question: #1 **History:** New for Bwd 2016 NRC exam RO level High Cog **K/A:** 002 RCS K5.10 Knowledge of the operational implications of the following concepts as they apply to the RCS: Relationship between reactor power and RCS differential temperature. RO 3.6 SRO 4.1 The question meets the K/A because the candidate must know the relationship of power to ΔT. TIER: 2 GROUP: 2 Task No: R-GP-027 Obj No: T.GP03-05 **10CFR55 Link:** 10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics. **Technical Reference with Revision Number:** Big note RC-3 rev 8, **Answer Explanation:** The Bwd Unit -1 DeltaT range is 0º to 60ºF. Because power is proportional to and changes linearly with DeltaT, 30ºF to 45ºF are 50% and 75% of full power DeltaT. Choice A is incorrect, see explanation above. This answer represents a ΔT range of 30ºF which is plausible because Bwds Tave range is 30ºF. Choice B is correct, see explanation above. Choice C is incorrect, see explanation above. This answer represents a ΔT range of 100ºF which is plausible because Bwds DeltaT meters read in percent power from 0 to 100%. Choice D is incorrect, see explanation above. This answer represents a ΔT range of 132ºF which is plausible because Bwds OTDT setpoint is 132ºF.

2 **ID: 1267508** **Points: 1.00**

Given:
- Unit 1 was at 100% power, normal alignment.
- Subsequently, Power Range N-41 failed HIGH.
- Currently the RPS bistables for N-41 are in bypass.
- The SM has directed the bistables be removed from bypass AND placed in trip to comply with administrative requirements.

With the above conditions, when N-41 bistables are removed from bypass AND placed in tripped, the coincidence logic for Power Range NI RPS actuation will change from __(bypassed)__ to __(tripped)__.

	(bypassed)	(tripped)
A.	1 of 3	2 of 3
B.	2 of 3	1 of 3
C.	2 of 3	3 of 3
D.	3 of 3	2 of 3

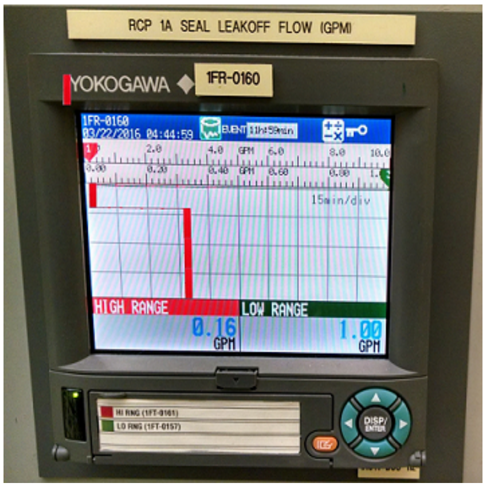
Answer: B

Answer Explanation
Bwd 2016 NRC Exam Question: #2 **History:** New for Bwd 2016 NRC exam RO level High Cog **K/A:** 015 Nuclear Instrumentation K3.01 Knowledge of the effect that a loss or malfunction of the NIS will have on the following: RPS. RO 3.9 SRO 4.3 The question meets the K/A, requires examinee knowledge of effect that a loss of NIs has on RPS. When an NI channel fails, the channel is typically placed in bypass until such time as the channel is repaired and placed back in service, or the Tech Spec time clock requires it to be placed in trip. TIER: 2 GROUP: 2 Task No: R-OA-055 Obj No: S-NI3-05-A through G **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** big note NI-2 rev 9, SSPS-1 rev9, lesson plan I1-NI-XL-03 **Answer Explanation:** Normal RPS actuation logic for the PRNIs is 2/4. When NI bistables are in bypass, the coincidence changes to 2/3 operable channels (because the bypassed channel SSPS input relays are energized regardless of bistable status). When the bistables are subsequently tripped, SSPS input relays are opened (de-energized regardless of bistable status). Therefore the coincidence changes to 1/3 operable channels. 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.

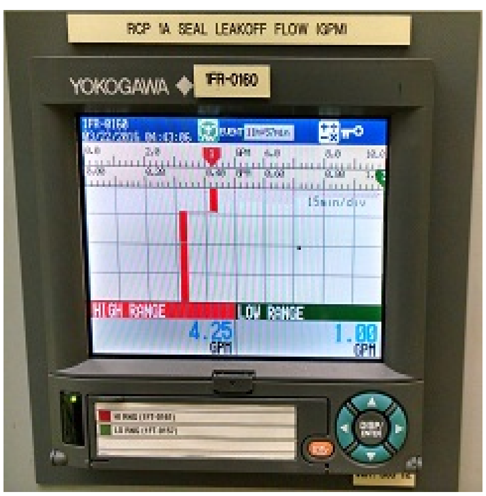
3 **ID: 1267570** **Points: 1.00**

Given:
- Unit 1 is at 99.7% power, normal alignment.
- The crew has recently established excess letdown flow per BwOP CV-15, EXCESS LETDOWN OPERATIONS.

With the above condition, which of the chart recorders below indicate the EXPECTED plant response to establishing excess letdown flow?

A-


B-


C-


D-


A. A

B. B

C. C

D. D

Answer: B

Answer Explanation
Bwd 2016 NRC Exam Question: #3 **History:** New for Bwd 2016 NRC exam RO level High Cog **K/A:** 016 Non-Nuclear Instrumentation A4.02 **Ability to manually operate and/or monitor in the control room**: Recorders. RO 2.7 SRO 2.6 The question meets the K/A, requires examinee to evaluate recorder for proper response to changing plant condition. TIER: 2 GROUP: 2 Task No: R-CV-007 Obj No: S.CV1-05-I **10CFR55 Link:** 10CFR55.41(b)5) Observe and safely control the operating behavior characteristics of the facility. **Technical Reference with Revision Number:** BwOP CV-15 rev 14 page 2, Big note CV-1 rev 15 **Answer Explanation:** When aligning excess letdown to the operating unit, a slight drop (approx. 1 gpm per RCP) in RCP seal return flow is expected because the excess letdown line intersects with the seal return line. This raises backpressure on the seal leakoff line and slightly lowers flow. Choice A is incorrect, This represents a near total loss of seal flow. Choice B is correct, see explanation above. Choice C is incorrect, This represents a rise of approx. 1 gpm per RCP. Choice D is incorrect, This represents a rise to approx. the seal flow that would represent a failed RCP seal.

EXAMINATION ANSWER KEY

2016 NRC

4	ID: 1267589	Points: 1.00

Given:

- The spent fuel pool level is as indicated below.
- The crew has recently swapped FC cooling pumps to support maintenance.
- The newly started FC cooling pump develops a large leak between the pump and discharge check valve.
- Spent Fuel Pool level is dropping and radiation levels at the pool surface are rising.

With the above conditions and NO operator action, which of the following is the approximate INDICATED water level that the Spent Fuel Pool will stabilize at?

A. 1' - 2'

B. 4' - 5'

C. 17' - 18'

D. 23' - 24'

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #4 **History:** Modified from Bwd systems (BWLI-FC1-019) RO level Low Cog **K/A:** 033 Spent Fuel Pool Cooling K4.04 Maintenance of spent fuel pool radiation. RO 2.7 SRO 2.9 The question meets the K/A because the candidate must know how far spent fuel pool level will drop (and therefore maintain water shielding of radiation in the pool) if a FC pump discharge line leak occurs. TIER: 2 GROUP: 2 Task No: Obj No: S.FC1-05 **10CFR55 Link:** 10CFR55.43(b)(7) Fuel handling facilities and procedures. Although this is an SRO item, the ROs have indication and ability to monitor SFP level in the MCR. **Technical Reference with Revision Number:** Big note FC-1 rev 12, M-63 sht 1A rev BA **Answer Explanation:** The suction piping of the FC cooling pumps is approx. 7' below the normal level (24' 6") of the SFP. 24' 6" - 7' = 17' 6" Choice A is incorrect. This is how far level would drop if both the transfer canal and cask fill area flooded from empty with 1FH001 open. Choice B is incorrect. This is the depth of the cooling water return lines were a break in the FC system downstream of the cooling pump discharge check valves siphon the pool. Choice C is correct. Choice D is incorrect. This is the approximate tech spec low level limit.

5 **ID: 1268888** **Points: 1.00**

Given:

- A Unit 1 xenon free reactor startup is in progress from a maintenance outage at MOL.
- The reactor was taken critical by withdrawing control rods.
- Reactor power was stable at 10^{-8} amps in the intermediate range for recording critical data.
- The RO then withdraws control rods 10 steps.

With the above conditions, reactor power will rise...

A. continuously above Mode 1 UNLESS control rods are manually inserted.

B. and stabilize at a value above 10^{-8} amps AND below the POAH.

C. temporarily, THEN drop back to 10^{-8} amps.

D. and stabilize at a value above the POAH AND below Mode 1.

Answer: D

Answer Explanation
Bwd 2016 NRC Exam Question:#5 **History:** Bank Bwd NOPS RO level High Cog **K/A:** 001 Control Rod Drive System A1.06 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CRDS controls including: Reactor power RO 4.1 SRO 4.4 The question meets the K/A because the candidate must know the relationship of power to ΔT. TIER: 2 GROUP: 2 Task No: R-GP-007 Obj No: 4B.GP-02 **10CFR55 Link:** 10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics. **Technical Reference with Revision Number:** LP I1-RT-XL-08 rev 10a page 12 **Answer Explanation:** When control rods are withdrawn from a stable critical condition below the POAH, a positve startup rate will result until negative reactivity is inserted to stabilize power. The negative reactivity will be from FTC and MTC when the POAH is reached and temperature feedback is received. This will result in reactor power stabilizing slightly above the POAH (2%-3%) reactor power. Mode 1 is 5% reactor power, so the reactor will stabilize below Mode 1. 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.

6	ID: 1267590	Points: 1.00

Given:

- Unit 1 is at 100% power, normal alignment.
- An alarm on the RM-11 indicates 0PR16J, SG Blowdown After Filter Outlet Rad Monitor has turned from Green to RED.

With the above condition, the...

A. 1SD02PA/B, Blowdown Condenser Hotwell Pumps, TRIP.

B. 1SD007, SG Blowdown Condenser Inlet Valve, CLOSES.

C. 1PS179A-D, SG Blowdown Sample Isolation Valves, CLOSE.

D. 0WX119A, CST Inlet Header Isolation from Blowdown Demin 0A Valve, CLOSES.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #6
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: 035 SGS
K1.11 Knowledge of the physical connections and/or cause-effect relationships between the S/GS and the following systems: PRM-systems.
RO 3.1 SRO 3.1

The question meets the K/A because the candidate must know the interlock of the process rad monitors that monitor SG blowdown .

TIER:	2
GROUP:	2
Task No:	R-AR-011
Obj No:	S.WX1-05-A

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Technical Reference with Revision Number: Big note AR-1 rev.10

Answer Explanation: A high radiation signal from 0PR016J will swap SG blowdown flow from the main condenser to the SG blowdown monitor tanks by closing 0WX119A and opening 0WX058A.
Choice A is incorrect. This is an interlock for low condenser level or high demin inlet pressure.
Choice B is incorrect. This is an interlock for high blowdown condenser pressure
Choice C is incorrect. This is an interlock for high radiation in the 1PR008J, SG Blowdown sample flow rad monitor.
Choice D is correct, see explanation above.

7	ID: 1267612	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 69% power.

D. begin to OPEN at approximately 74% power.

Answer: B

Answer Explanation
Bwd 2016 NRC Exam Question: #7 **History:** Bank from 2009 Bwd NRC exam RO level High Cog **K/A:** 041 Steam Dump System A3.05.Ability to monitor automatic operation of the SDS, including: Main steam pressure RO 2.9 SRO 2.9 The question meets the K/A, requires examinee ability to monitor changing steam pressure and predict impact of on steam dumps controls. TIER: 2 GROUP: 2 Task No: R-DU-001 Obj No: S-DU1-03-A **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** big note MS-4, ILT lesson plan I1-DU-XL-01 rev. 6a page 7 **Answer Explanation:** 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, but plausible. Because steam pressure rises as power drops, a commom error would be to interpolate from the wrong end of the steam band resulting in an answer of 39% (<50%). Choice B is correct. see explanation above. Choice C is incorrect, but plausible.Using steam values of 1115 psig (SG PORV opening setpoint) and 1005 psig, the same calculation results in an answer of 69%. Choice D is incorrect, but plausible. Using U-2 steam values of 1092 psig and 885 psig, the same calculation results in an answer of 74%.

8	ID: 1267628	Points: 1.00

Given:
- U-1 was at 100% power normally aligned.

The following events then occur:
- A reactor trip occurs.
- Bus 144 Auto Bus Transfer fails and the Bus is DE-ENERGIZED.
- The crew has cross tied bus 144 to Bus 142 and re-energized bus 144.

With the above conditions, which of the following PREVIOUSLY RUNNING bus 144 loads will require local operator action at the 4kv breaker to re-start the load?

A. Circulating Water Pump

B. Non-Essential Service Water Pump.

C. Station Air Compressor

D. Containment Chiller

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #8 **History:** New for Bwd 2016 NRC exam RO level High Cog **K/A:** 079 Station Air System (SAS) G2.1.30 Ability to locate and operate components, including local controls RO 4.4 SRO 4.0 The question meets the K/A because the candidate must know that the SAC feed breakers are unique loads on the non-ESF 4kv busses because they have 86 lockout relays that must be manually/locally reset after a bus under voltage. TIER: 2 GROUP: 2 Task No: R-SA-001 Obj No: S.SA1-1-02-A **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** ILT Lesson plan I1-SA-XL-01 rev. 6 page 36, BwOP SA-1 rev 043 page 6 **Answer Explanation:** SAC feed breakers are unique loads on the non-ESF 4kv busses because they have 86 lockout relays that must be locally reset after a bus under voltage. All three distractors are bus 144 4kv loads but do not have lockout relays. This was a past OPEX at Braidwood. After aloss of offsite power on Unit 1 while the unit was shutdown, the associated SAC feed breakers were unavailable to auto start for approx. a day after the bus power was restored. Choice A is incorrect, although CW pumps have field monitoring relays, they are not intended to trip in an undervoltage condition as described. Choice B is incorrect, see explanation above. Choice C is correct, see explanation above. Choice D is incorrect, see explanation above.

9	ID: 1267630	Points: 1.00

Given:

- U-1 was at 100% power, normally aligned.
- The 1A SX Pump is OPERATING.

The following then occurs:
- DC Bus 111 is FAULTED and DE-ENERGIZES.
- 15 seconds later, the reactor is MANUALLY tripped.
- 30 seconds after the reactor trip, A LOSS OF OFF SITE (LOOP) power occurs.

With the above conditions, 40 seconds after the LOOP, which (if any) SX pps will be OPERATING?

A. BOTH 1A and 1B SX pumps will be SHUTDOWN.

B. BOTH 1A and 1B SX pumps will be OPERATING.

C. 1A SX pump will be OPERATING and 1B SX pump will be SHUTDOWN.

D. 1B SX pump will be OPERATING and 1A SX pump will be SHUTDOWN.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #9
History: New for Bwd 2016 NRC exam
RO level
High Cog

K/A: 075 Circulating Water System
K2.03Knowledge of bus power supplies to the following: Emergency/essential SWS pumps
RO 2.6 SRO 2.7

The question meets the K/A because the candidate must know the SX pump power supplies and the breaker control power supplies.

TIER:	2
GROUP:	2
Task No:	R-SX-002
Obj No:	S.SX1-05-C

10CFR55 Link: 10CFR55.41(b)(5)

Technical Reference with Revision Number: Big Note AC-7 rev. 8. and Big Note DG-6 rev. 9. Drawing 20E-1-4030SX01 rev. U. Drawing 20E-1-4030SX02 rev. V.

Answer Explanation: When DC bus 111 faults, The 1A SX pump will lose control power to it's breaker in the closed position. When the LOOP occurs, Bus 141 will de-energize and will not re-energize because the 1A DG will not start nor re-energize bus 141 due to the loss of DC bus 111. Therefore, even though the 1A SX pump breaker is closed, the pump will not re-energize. The 1B SX pump will start on the LOOP DG sequencer 25 seconds after the 1B DG re-energizes bus 142.
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.

10 **ID: 1267632** **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 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
Bwd 2016 NRC Exam Question: #10 **History:** Bank from 2009 Bwd NRC exam RO level Low Cog **K/A:** 086 Fire Protection K6.04 Knowledge of the effect of a loss or malfunction on the Fire Protection System will have on the following: Fire, smoke, and heat detectors. RO 2.6 SRO 2.9 The question meets the K/A, requires examinee ability to monitor the bypass of a fire zone detector. TIER: 2 GROUP: 2 Task No: R-FP-002 Obj No: S-FP1-08 **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features **Technical Reference with Revision Number:** Big Note FP-2 rev. 7 **Answer Explanation:** 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 1PM09J 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

11 **ID: 1267954** **Points: 1.00**

Given:

- Unit 1 is at 100% power, normal alignment.
- An inadvertent Phase A actuation occurs.

With the above conditions, the RCP #1 seal leakoff flows are directed to the...

A. Pressurizer Relief Tank.

B. Reactor Coolant Drain Tank.

C. Containment Floor Drain Sump.

D. Containment Reactor Cavity Sump.

Answer: A

Answer Explanation
Bwd 2016 NRC Exam Question: #11 **History:** Modified from Bwd systems (BWLI-RC2-033) RO level Low Cog **K/A:** 003 RCPS K6.04 Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Containment isolation valves affecting RCP operation RO 2.8 SRO 3.1 The question meets the K/A because the candidate must know the plant affects if RCP seal leakoff containment isolation valves fail closed. TIER: 2 GROUP: 1 Task No: Obj No: **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features **Technical Reference with Revision Number:** Big Note CV-1 rev. 15. **Answer Explanation:** When a phase A signal actuates, the RCP seal leakoff containment isolation valves 1CV8100 and 1CV8112 close. The seal leakoff path back to the VCT is isolated. A 150# relief valve in the seal leakoff line will lift and redirect the flow to the PRT. Choice A is correct, see explanation above. Choice B is incorrect but plausible because the #2 RCP seals are normally directed to the RCDT. Choice C is incorrect but plausible because the #3 RCP seals are normally directed to the containment floor drain sump. Choice D is incorrect, but plausible because as explained above, the #3 seals normally drain to the containment floor drain sump and the reactor cavity sump is commonly confused with it. Also, the reactor cavity sump is pumped directly into the containment floor drain sump adding to the confusion.

12	ID: 1267955	Points: 1.00

Given:

- Unit 1 was at 100% power, normal alignment.

The following occurs:

- Bus 132X faults and is DE-ENERGIZED.
- Simultaneously, an RCS LOCA occurs causing a Reactor Trip and SI.

With NO operator actions, WHEN plant parameters meet the applicable low RWST level or low RCS pressure conditions, _______________ will FAIL to stroke.

A. 1MOV-CV8110, 1B CV PP MINIFLOW ISOL VLV

B. 1MOV-CV8111, 1A CV PP MINIFLOW ISOL VLV

C. 1SOV-CV8114, 1A CV PP MINIFLOW ISOL VLV

D. 1SOV-CV8116, 1B CV PP MINIFLOW ISOL VLV

Answer: B

Answer Explanation
Bwd 2016 NRC Exam Question: #12 **History:** New for Bwd 2016 NRC exam RO level Low Cog **K/A:** 004 CVCS K2.05 Knowledge of bus power supplies to the following: MOVs RO 2.7 SRO 2.9 The question meets the K/A because the candidate must know the power supplies to the respective CV pp miniflow isolation valves. TIER: 2 GROUP: 1 Task No: R-CV-006 Obj No: S.CV1-16-C **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features **Technical Reference with Revision Number:** 20E-1-4030CV16 rev. O. **Answer Explanation:** 1CV8110 and 1CV8111 have "cross train" power supplies. The 1A CV pp miniflow valve, 1CV8111 has a division 12 power supply (MCC 132X4) and likewise for 1CV8110 for the 1B CV pp which has a division 11 power supply (MCC131X1). Whereas, 1CV8114 and 8116 are powered from the same division as the CV pp they serve (from DC system). This is a design feature to prevent a CV pp from not isolating during a LOCA (if a single train of 480VAC or 125VDC power fails) and spread highly radioactive water to the aux bldg during ECCS recirc phase. 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.

13	ID: 1267958	Points: 1.00

Given:

- The Unit 1 RO is re-aligning the 1A RH pump from shutdown cooling to ECCS cold leg injection.
- The RO is about to OPEN 1SI8812A, PP 1A SUCT FROM RWST ISOL VLV.

With the above conditions, which of the following is an interlock that must be met to OPEN 1SI8812A?

A. 1CS001A, PP 1A RWST SUCT VLV must be OPEN.

B. 1CV8804A, PP 1A SUMP SUCT VLV must be OPEN.

C. 1SI8811A, CNMT SUMP 1A ISOL VLV must be CLOSED.

D. 1RH8701A, RC LOOP 1A TO RH PP 1A SUCT ISOL VLV must be CLOSED.

Answer: C

Answer Explanation

Bwd 2016 NRC Exam Question: #13
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: 005 RHRS
K4.02 Knowledge of RHRS design feature(s) and/or interlock(s) which provide for the following: Modes of operation
RO 3.2 SRO 3.5

The question meets the K/A because the candidate must know the interlocks of RH system valves required to change the system mode of operation.

TIER:	2
GROUP:	1
Task No:	R-EF-013
Obj No:	S.EC1-07-A

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Technical Reference with Revision Number: Big note ECCS-2 rev.10.

Answer Explanation: The 1SI8812A valve is interlocked such that 1SI8811A and 1CV8804A must both be closed to enable manual opening of 1SI8812A.
Choice A is incorrect, 1CS001A is interlocked with 1SI8811A (not 1SI8812A).
Choice B is incorrect, 1CV8804A must be closed (not open).
Choice C is correct, see explanation above.
Choice D is incorrect, 1RH8701A is interlocked such that 1SI8812A must be closed to open 1RH8701A (the opposite).

14 **ID: 1267960** **Points: 1.00**

Given the following sequence of events:

- Unit 1 is at 100% power with pressurizer pressure channel, 1P-455 in test (bistables tripped).
- ALL other equipment is in a 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 RO 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
Bwd 2016 NRC Exam Question: #14 **History:** Bank from Bwd 2009 NRC Exam RO level High Cog **K/A:** 006 Emergency Core Cooling K4.21 Knowledge of ECCS design feature(s) and/or interlock(s) which provide for the following: Bypassing/blocking ESF channels RO 4.1 SRO 4.3 The question meets the K/A because the candidate must know actions that will block automatic SI actuation. TIER: 2 GROUP: 1 Task No: R-EP-036 Obj No: S.EF1-06 **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features **Technical Reference with Revision Number:** Lesson plan I1-EF-XL-01 rev. 5. page 19 BwAR 1-BP-5.1 rev 0. **Answer Explanation:** 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 and 1PT-456 & 457 will be above P-11. Choice C is incorrect, this light will be lit when pressure 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.

15 **ID: 1267961** **Points: 1.00**

Given:

- Unit 1 is at 100% power, normal alignment.
- A slow and steady rise in PRT level has been noted over several hours.
- Annunciator 1-12-A7, PRT LEVEL HIGH/LOW has just alarmed.

With the above conditions, the PRT level will be lowered by …

A. verifying 1RE1003, RCDT Pumps Discharge Cnmt Isol Valve, auto opens on high PRT level, then verify the 1A RCDT pump auto starts.

B. verifying 1RE1003, RCDT Pumps Discharge Cnmt Isol Valve, auto opens on high PRT level, then verify the 1B RCDT pump auto starts.

C. manually opening 1RY8031, PRT Drain Isol Valve, and 1RE1003, RCDT Pumps Discharge Cnmt Isol Valve, then verifying the 1A RCDT pump auto starts.

D. manually opening 1RY8031, PRT Drain Isol Valve, and 1RE1003, RCDT Pumps Discharge Cnmt Isol Valve, then verifying the 1B RCDT pump auto starts.

Answer: D

Answer Explanation
Bwd 2016 NRC Exam Question: #15 **History:** Bank from 2014 Bwd NRC exam RO level Low Cog **K/A:** 007 Pressurizer Relief Tank/Quench Tank System (PRTS) A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRTS controls including: Maintaining quench tank water level within limits RO 2.9 SRO 3.1 The question meets the K/A because examinee must be able to monitor changes in parameters associated with the PRT water level. TIER: 2 GROUP: 1 Task No: R-EP-036 Obj No: 3D.EP-02-E **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features **Technical Reference with Revision Number:** Big note RY-4 rev. 9, 20E-1-4030RE01. **Answer Explanation:** When 1RE1003, RCDT Pumps Discharge Containment Isolation Valve and 1RY8031, PRT Drain Valve, are manually opened, the 1B RCDT pump will auto start. A is incorrect. Plausible since at 60% Level in the RCDT 1A pump will auto start. B is incorrect. Plausible since at 80% Level in the RCDT 1B pump will auto start. C is incorrect. Plausible if the examinee thinks the RCDT 1A pump will start when 1RY8031 is opened. D is correct. See explanation.

16 **ID: 1267962** **Points: 1.00**

Given:

- Braidwood Station is in a dual unit outage.
- Both Units are in Mode 4.
- The U-0 Component Cooling HX is aligned to Unit 2.
- The 1A RH Train is in Shutdown Cooling Mode.
- The 2B RH Train is in Shutdown Cooling Mode.
- The 1B & 2A RH Trains are aligned for ECCS cold leg injection.

The MCR crew desires to raise SX flow to maximize cooldown rate capabilities.

With the above conditions, throttling OPEN 0SX007, U-0 SX OUTLET FROM CC HX VLV, will raise cooldown rate capabilities for...

A. BOTH Units.

B. ONLY Unit 1.

C. ONLY Unit 2.

D. NEITHER unit.

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: #16
History: New for 2016 Bwd NRC exam
RO level
High Cog

K/A: 008 Component Cooling Water System (CCWS)
K1.01 Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: SWS
RO 3.1 SRO 3.1

The question meets the K/A because examinee must know the CC system alignment and how it is affected by the SWS, and affects the RH system.

TIER:	2
GROUP:	1
Task No:	R-CC-002
Obj No:	S.CC1-02-B

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Technical Reference with Revision Number: Big Note CC-1 rev 16, M-66 series, Sheet 4 D rev BC.

Answer Explanation: With the U-0 CC HX aligned to U-2, U-2 CC system will be supplying the A train RH loop on both units. Each B train RH system will be supplied by the respective units CC system. Therefore adjusting SX flow to the U-0 CC HX will raise cooldown capabilities of both units RH systems in the alignment given.
A is correct. See explanation.
B is incorrect. this would be true if U-0 Hx was aligned to U-1. See explanation.
C is incorrect. See explanation.
D is incorrect. See explanation. Neither unit is plausible because CC can be aligned in post LOCA cooldown configuration with 1B and 2B RH trains in service. In this configuration, neither units cool down rate would be affected by 0SX007 position.

17	ID: 1267968	Points: 1.00

Given:

- Unit 1 is at 50% power, normal alignment.
- PZR Pressure Control Channel Select switch is in the "PT-455/PT-456" position.

The following occurs:

- 1PK-455A, Master PZR Pressure Controller "potentiometer" setting fails from its normal setting to the equivalent of 3.0.

With the above conditions and NO operator action, RCS pressure will INITIALLY...

A. LOWER to the low PZR pressure reactor trip setpoint.

B. LOWER and stabilize ABOVE the low PZR pressure reactor trip setpoint.

C. RISE to the high PZR pressure reactor trip setpoint.

D. RISE and PZR pressure will be controlled near 2315-2335 psig by a PZR PORV.

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #17
History: Bank from Bwd LORT
RO level
High Cog

K/A: 010 Pressurizer Pressure Control
K1.03 Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: RCS
RO 3.6 SRO 3.7

The question meets the K/A because the candidate must know how a malfunction in the PZR PCS affects the actual RCS pressure.

TIER:	2
GROUP:	1
Task No:	R-OA-100
Obj No:	S.RY1-21-E

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Technical Reference with Revision Number: Big note RY-2 rev. 10.

Answer Explanation: The normal PZR master controller auto setting is 6.688 (on a scale of 1-10, 1700#-2500#) representing a setpoint of 2235 psig. If the controller fails to an equivalent of 3.0, it will try to control pressure at 1940 psig, which is above the low press Rx trip setpoint of 1885 psig. The distractors of pressure rising would be plausible for a reverse acting controller and the pressures are the PZR PORV cycling setpoint (2335#-2315#) and high reactor trip setpoint.
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.

18	ID: 1267969	Points: 1.00

Given:

- Unit 1 was at 100% power, normal alignment.
- Subsequently, a loss of DC bus 111 occurs.
- The crew initiates a manual reactor trip and enters 1BwEP-0, REACTOR TRIP OR SI.

Note:

- "TSLB-4" is the NIS AND CONTAINMENT TRIP STATUS LIGHTS at 1PM05J.
- "Position indication lights" are the red and green "checkerboard" lights at 1PM05J.

With the above conditions, the position of RTA (train A reactor trip breaker) can be obtained locally...

A. and from BOTH TSLB-4 and its position indication lights.

B. and from its position indication lights (but NOT from TSLB-4).

C. and from TSLB-4 (but NOT from its position indication lights).

D. but NOT from either its position indication lights OR TSLB-4 .

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #18 **History:** Bank from Bwd 2011 NRC Exam RO level Low Cog **K/A:** 012 Reactor Protection System (RPS) K1.02 Knowledge of the physical connections and/or cause effect relationships between the RPS and the following systems: 125V DC system RO 3.4 SRO 3.7 Question meets K/A, requires examinee knowledge of how the loss of DC bus will affect the RPS indications. TIER: 2 GROUP: 1 Task No: R-EP-011 Obj No: 3D.EP-01-A **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** 20E-1-4030RD06, Rev S. **Answer Explanation:** Power to the position indication lights for RTA come from the reactor trip breaker control circuit (DC bus 111). Therefore the position indication lights will not work. However, the TSLB-4 lights are powered through SSPS which relies on instrument bus power and will continue to work properly even with a loss of the DC bus 111. 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.

19 **ID: 1267988** **Points: 1.00**

Given:

- Unit 1 is at 100% power.
- The operating crew is performing 1BwOSR 3.3.1.4-2, UNIT ONE SSPS, REACTOR TRIP BREAKER, AND REACTOR TRIP BYPASS BREAKER SURVEILLANCE with the following conditions:

  - BOTH Reactor Trip Breakers (RTA and RTB) are CLOSED.
  - 1B Reactor Trip Bypass Breaker (BYB) is RACKED IN and CLOSED.
  - 1B Input Error Inhibit switch is in INHIBIT.
  - 1B Mode Selector switch is in TEST.
  - 1B Multiplexer Test switch is in INHIBIT.
  - 1A Multiplexer Test switch is in NORMAL.

All other systems are normally aligned, then:

- Pressurizer Pressure drops below the automatic SI setpoint.

Based on the above conditions and assuming NO operator actions are taken ...

A. SI actuation will occur on ONLY the 1A SSPS train, AND ONLY RTA will OPEN (RTB and BYB will remain CLOSED).

B. SI actuation will NOT occur on either SSPS train, AND RTA, RTB, and BYB will ALL remain CLOSED.

C. SI actuation will occur on ONLY the 1A SSPS train, AND ONLY RTA and BYB will OPEN (RTB will remain CLOSED).

D. SI actuation will occur on BOTH SSPS trains, AND RTA, RTB, and BYB will ALL OPEN.

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #19 **History:** Bank from 2014 Backup Cert Exam RO level High Cog **K/A:** 013 Engineered Safety Features Actuation System (ESFAS) K5.02 Knowledge of the operational implications of the following concepts as they apply to the ESFAS: Safety system logic and reliability RO 2.9 SRO 3.3 The question meets the K/A because the candidate must know how safety system logic and reliability changes during testing. TIER: 2 GROUP: 1 Task No: R-EF-013 Obj No: S.EF1-08 **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** Lesson Plan I1-RP-XL-01 rev. 5b, page 19 and Big Note EF-1 rev. 15. **Answer Explanation:** The 1B input error inhibit switch in INHIBIT prevents a logic ground for any logic input on train B. This in turn prevents the SI from actuating on train B. The train B SSPS mode switch in test prevents train B reactor trip. The RTA and BYB receive a reactor trip signal from train A 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.

20	ID: 1267989	Points: 1.00

Given:

- Unit 1 is at full power.
- 1A, 1B and 1C RCFCs are running in high speed with the 1D RCFC in standby.
- A reactor trip and Safety Injection occur due to high containment pressure.

After the SI occurs, what is the response of the RCFCs?

A. ALL RCFCs start immediately in low speed.

B. 1A, 1B and 1C RCFCs start immediately in low speed, and the 1D RCFC starts 20 seconds later in low speed.

C. 1A, 1B and 1C RCFCs start 20 seconds later in low speed, and the 1D RCFC starts immediately in low speed.

D. ALL RCFCs start 20 seconds later in low speed.

Answer: D

Answer Explanation
Bwd 2016 NRC Exam Question: #20 **History:** Bank from Bwd LORT RO level Low Cog **K/A:** 022 Containment Cooling System K4.02 Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: Correlation of fan speed and flowpath changes with containment pressure RO 3.1 SRO 3.4 The question meets the K/A because the candidate must know interlocks for RCFC fans speed changes. TIER: 2 GROUP: 1 Task No: R-VP-004 Obj No: S.VP1.08-A **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** 20E-1-4030VP01 rev. O. (Also VP03,VP05 and VP07 also show same TDR for those respective RCFC low speed) **Answer Explanation:** The RCFC low speed breaker control circuit contains a 20 second time delay relay which is energized whenever an auto start signal (SI) is generated. The time delay relay contact is only in the standby (auto) start path of the circuit. If a manual start was attempted of an RCFC low speed breaker, there is no time delay creating plausibility for the distractors. 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.

21	ID: 1268008	Points: 1.00

Given:

- Unit 2 has experienced a reactor trip and SI from full power due to an RCS LOCA.
- 2SI8811A, CNMT SUMP 2A ISOL VLV, failed to automatically open when the appropriate plant conditions exist.
- ALL other plant equipment operates as expected.
- The crew is currently performing 2BwEP ES-1.3, TRANSFER TO COLD LEG RECIRC, Attachment A, MANUAL OPERATION OF CONTAINMENT SUMP ISOLATION VALVES.
- 2SI8811A was JUST manually opened and the 2A RH Pump and the 2A CS Pump were restarted.
- 2CS009A, PP 2A SUMP SUCT VLV, is CLOSED.
- RWST level is 8% and slowly lowering.

With the above conditions, the crew will ________(1)________ because ________(2)_________.

A. (1) ALLOW the 2A CS pump to continue running
(2) RWST is still ABOVE the required level

B. (1) ALLOW the 2A CS pump to continue running
(2) a flowpath from the ECCS sump currently EXISTS

C. (1) PLACE the 2A CS pump in PTL
(2) RWST is BELOW the required level AND NO flowpath from the ECCS sump currently exists

D. (1) PLACE the 2A CS pump in PTL
(2) NO flowpath from the ECCS sump NOR the RWST currently exists

Answer: C

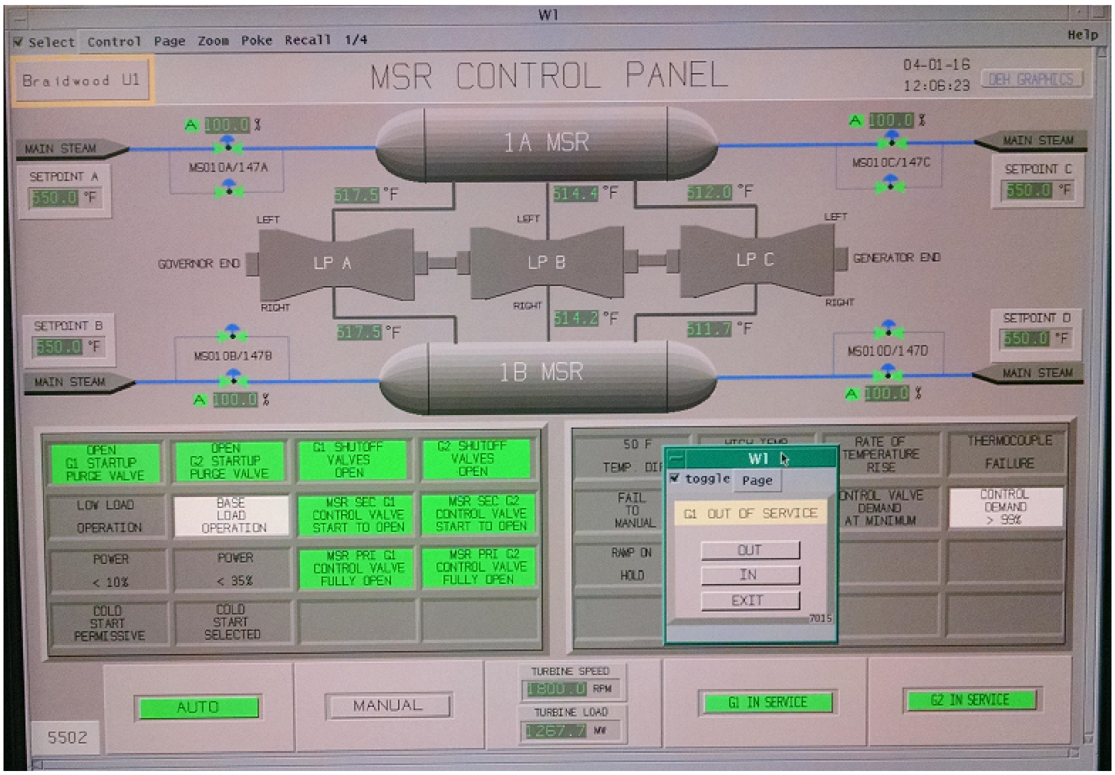
Answer Explanation
Bwd 2016 NRC Exam Question: #21 **History:** New for Bwd 2016 NRC exam RO level High Cog **K/A:** 026 Containment Spray System A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic recirculation transfer RO 4.2 SRO 4.4 The question meets the K/A because the candidate must know the 2A CS pump suction alignment after performing 2BwEP ES-1.3 Attach A. and must know the mitigation strategy on the OAS page. TIER: 2 GROUP: 1 Task No: R-CS-003 Obj No: S.CS1-08-B **10CFR55 Link:** 10CFR55.41(b)(8) Components, capacity, and functions of emergency systems. **Technical Reference with Revision Number:** Big Note CS-1 rev. 15 and 2BwEP ES-1.3 rev. 204 OAS page. **Answer Explanation:** Because the 2SI8811A did not auto open when required, the crew would have to perform manual actions in 2BwEP ES-1.3 Attach A, which would delay the time that CS would normally be manually aligned to the containment sump. This delay causes RWST level to drop below the required level for continued CS pump suction supply. Per the 2BwEP ES-1.3 OAS page, IF RWST level drops to 9%, stop any pump with a suction flowpath from the RWST and realign the pump to the ECCS recirc alignment. To align the 2A CS pump to the sump, the operator would have to close 2CS001A and open 2CS009A. Choice A is incorrect, 2A CS pump must be placed in PTL per OAS. Choice B is incorrect, 2A CS pump must be placed in PTL per OAS. Choice C is correct, see explanation above. Choice D is incorrect, suction alignment from the RWST currently exists, just not enough level in RWST to support pump operation.

22 **ID: 1268009** **Points: 1.00**

Given:

- Unit 1 is at 50% power, normal alignment.
- The RO is about to initiate an MSR OOS.
- The G1 IN SERVICE poke field was clicked on (see MSR control panel below).

When the RO clicks on the "OUT" poke field in the pop-up window, which of the following valves are expected to automatically CLOSE?



A. 1MS010A&B AND 1MS147A&B (only)

B. 1MS010A&C AND 1MS147A&C (only)

C. 1MS010A&D AND 1MS147A&D (only)

D. 1MS010A,B,C&D (only)

Answer: A

Answer Explanation
Bwd 2016 NRC Exam Question: #22 **History:** Bank from Bwd Systems RO level Low Cog **K/A:** 039 A3.02 Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS RO 3.1 SRO 3.5 The question meets the K/A because the candidate must know the auto functions of the MSR controller to properly monitor operations. TIER: 2 GROUP: 1 Task No: R-MS-014 Obj No: S.MT1-08 **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** Lesson plan I1-MT-XL-01 rev. 5a, page 24. Big note MS-3 rev 12. **Answer Explanation:** Each MSR has two sets of tube bundles (one at each end). When MSR tube bundles are taken OOS at power, they are ramped off in pairs (one tube bundle from each side of the main turbine) regardless of which bundle needs maintenance. This is to provide even heating across the main turbine and prevent asymetrical expansion or contraction of the turbine system. Therefore, when the G1 (group 1) tube bundles are taken OOS it ramps closed the A and B steam supply valves. The RO must have knowledge of this configuration to properly monitor the evolution. This configuration is easily confused with other plant equipment that is grouped by trains in either A/D and B/C configuration or by the entire MSR A/C and B/D configuration. Choice A is correct, see explanation above. Choice B is incorrect, this configuration would remove both tube bundles from the 1A MSR and nothing from the 1B MSR. Choice C is incorrect, this would remove tube bundles at opposite ends of the 1A and 1B MSRs. Choice D is incorrect, this would close the large steam supply valve (1MS010s) but not the small steam supply valves (1MS047s).

23	ID: 1268010	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.
- U-1 SG levels are 50% and slowly dropping.

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

With the above conditions, the NEXT crew action is to...

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
Bwd 2016 NRC Exam Question: #23 **History:** Bank from Bwd 2009 NRC Exam RO level Low Cog **K/A:** 059 Main Feedwater (MFW) System A1.03 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 RO 2.7 SRO 2.9 The question meets the K/A, requires examinee knowledge of power restrictions of MFW pumps. TIER: 2 GROUP: 1 Task No: R-OA-004 Obj No: 4D.OA-20 **10CFR55 Link:** 10CFR55.41(b)(4) Secondary coolant and auxiliary systems that affect the facility. **Technical Reference with Revision Number:** 1BwOA SEC-1rev. 106, page 5, step 4b RNO **Answer Explanation:** 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 S/U FW pump. However, it is plausible because starting the strat up FW pump would add feed flow to the SG and is a success path for regaining a heat sink in emergency procedures. Choice D is correct, see explanation above.

24	ID: 1269468	Points: 1.00

Given :

- A U-2 plant heatup is in progress per 2BwGP 100-1, PLANT HEATUP.
- RCS temperature is 300°F and slowly rising.

With the above conditions, feed flow should be maintained in the...

A. main feed line to prevent the 2FW079A-D, FW FLOW CHECK VLV, from failing closed.

B. main feed line to prevent water hammer in the main feedwater nozzle.

C. tempering feed line to prevent water hammer in the auxilary feedwater nozzle.

D. main feed line to prevent water hammer in the steam generator feedwater preheater.

Answer: C

Answer Explanation

Bwd 2016 NRC Exam Question: #24
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: 061 Auxiliary/Emergency Feedwater (MFW) System
K5.05 Knowledge of the operational implications of the following concepts as the apply to the AFW: Feed line voiding and water hammer
RO 2.7 SRO 3.2

The question meets the K/A, requires examinee knowledge of the feed flow requirements to prevent water hammer in the auxilary feedwater nozzle.

TIER:	2
GROUP:	1
Task No:	R-GP-004
Obj No:	3B.GP-01-A-3

10CFR55 Link: 10CFR55.41(b)(4) Secondary coolant and auxiliary systems that affect the facility.

Technical Reference with Revision Number: 2BwGP 100-1 rev 032 page 59

Answer Explanation: Per 2BwGP 100-1, Limitation and Action E.7 "Whenever RCS temperature is >250°F, flow should be maintained through the Auxiliary Feed Nozzle, to prevent water hammer, which could result from intermittent flow".

Choice A is incorrect, but plausible because 2BwGP 100-3 contains a precaution that states "When feedwater temperature to the steam generators is less than 250°Fand the 2FW079 check valves are open, it is possible the check valves could fail closed as feedwater temperature drops. To minimize the potential for this occurrence do not hold reactor power between 2% and 12% unless permission from the Shift Manager has been received." In the above condition reactor power is not between 2% and 12%, so the condition is not applicable.
Choice B is incorrect, but plausible because there are interlocks of flow and purge permissives on the unit 2 main feedwater isolation valves designed to prevent main feedline water hammer conditions. However, these would not be applicable during a plant heatup.
Choice C is correct, see explanation above.
Choice D is incorrect, but plausible because there is a Water Hammer Prevention System (WHPS) on unit 2 that is designed to prevent SG preheater section water hammer. However, the system is based upon SG parameters vs. feedline flow.

25 **ID: 1268014** **Points: 1.00**

Given:

- Unit 1 is at full power, normal alignment with the following exception:
- 1BwOSR 3.8.1.2-1, 1A DIESEL GENERATOR OPERABILITY SURVEILLANCE, is in progress.
- The 1A DG is running and supplying bus 141 in parallel with the SATs.

The following then occurs:

- A Loss Of Offsite Power (LOOP) occurs due to a fault in SAT 142-1 actuating 86ST11A, SAT 142-1 Lockout Relay.

With the above conditions, breaker ACB 1413, DG 1A FEED TO 4KV BUS 141, will ________(1)________. After the event, the 1A DG will be operating in ______(2)_____ mode.

A. (1) REMAIN closed
(2) Droop

B. (1) REMAIN closed
(2) Isochronous

C. (1) TRIP open and RE-CLOSE
(2) Droop

D. (1) TRIP open and RE-CLOSE
(2) Isochronous

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #25
History: New for Bwd 2016 NRC exam
RO level
High Cog

K/A: 062 A. C. Electrical Distribution
K3.02 Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: ED/G
RO 4.1 SRO 4.4

The question meets the K/A, requires examinee knowledge of the effect of a AC distribution malfunction has on the DGs.

TIER:	2
GROUP:	1
Task No:	R-DG-011
Obj No:	S.AP1-11-C

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Technical Reference with Revision Number: ILT Lesson plan I1-AP-XL-01 rev. 4c, page 45-46, BwAR 1-21-A7 rev 6

Answer Explanation: When a DG is initially paralleled to the SATs, it is operating in droop mode to control DG load. A SAT fault will trip breaker 1412 (SAT feed to bus 141) open but not 1413 because it is only a trip of 1412 vs. a lockout of 1412 (which would trip 1413 open). Therefore 1413 remains closed through the event. Additionally, the trip of 1412 affects the DG control circuit by energizing relay 3IMX. This relay (when energized) places the DG in isochronous mode.

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.

26	ID: 1271429	Points: 1.00

Which of the following electrical busses supplies power to the 1TO05P, Main Turbine Emergency Oil Pump?

A. AC Bus 132X

B. MCC 133Z2

C. DC Bus 111

D. DC Bus 123

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #26
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: 063 D.C. Electrical Distribution
K2.01 Knowledge of bus power supplies to the following: Major DC loads
RO 2.9 SRO 3.1

The question meets the K/A by requiring knowledge of the power supply to a major DC load, the Main Turbine Emergency Oil Pump.

TIER:	2
GROUP:	1
Task No:	R-TO-004
Obj No:	S.DC2-02

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Technical Reference with Revision Number: 20E-1-4030TO04 rev. H

Answer Explanation: 1TO05P, Main Turbine Emergency Oil Pump is supplied by 250 VDC Bus 123. The 1TO05P is the second backup source of supplying oil to the main turbine in the event of a loss of normal supply by the main turbine shaft driven oil pump or the 1TO06P, Bearing Oil Pump (first backup).

Choice A is incorrect, but plausible because the 1TO06P Bearing Oil Pump is supplied by AC bus 132X.
Choice B is incorrect, but plausible because MCC 133Z2 powers other turbine non-ESF loads including one of the Main Oil Pumps to the 1C Feedwater Pump.
Choice C is incorrect, but plausible because the Main Turbine Emergency Oil Pump name implies it is an "emergency" load and DC bus 111 supplies multiple ESF or "emergency" loads.
Choice D is correct, see explanation above.

27 | **ID: 1268029** | **Points: 1.00**

Given:

- Unit 1 is at 100% power.
- A loss of 4KV Bus 142 has occurred.
- The 1B DG did NOT start.
- An EO is locally performing 1BwOA ELEC-3, Attachment D, LOCAL START OF 1B DG.
- When the DC Control Power Available lights are checked, they are noted to be NOT LIT.

Which of the following DC buses will the EO check that the associated breaker is closed for each indicating light on the 1B DG local control panel?

	DC POWER ON/BUS # 1 Light	DC POWER ON/BUS # 2 Light
A.	Bus 111	Bus 112
B.	Bus 112	Bus 111
C.	Bus 112	Bus 112
D.	Bus 112	Bus 114

Answer: C

Answer Explanation

Bwd 2016 NRC Exam Question: #27
History: Bank from 2011 Bwd NRC exam
RO level
Low Cog

K/A: 064 Emergency Diesel Generator (ED/G) System
Knowledge of bus power supplies to the following:
K2.03 Control power
RO 3.2 SRO 3.6

The question meets the K/A, requires examinee knowledge of DG control bus power supplies.

TIER:	2
GROUP:	1
Task No:	R-OA-013
Obj No:	S.DG1-03-D

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
.
Technical Reference with Revision Number: 20E-1-4030DG51 rev. AP, 20E-1-4030DG52 rev. AG

Answer Explanation:
Each DG has two control power circuits which supply different multiple functions of the DG auxiliaries. These circuits are NOT redundant (common misconception) and both are supplied from the DC ESF bus in the same division as the DG. For the 1B DG, each circuit is fed from a separate supply breaker on DC bus 112.
Choice A is incorrect, this would assume that circuit #1 is fed from bus 111 and circuit #2 is fed from bus 112 (misconception that the indicating light nomenclature is associated with respective DC ESF buses).
Choice B is incorrect, this would assume that circuit #1 is fed from bus 112 and circuit #2 is fed from bus 111 (misconception that the buses are redundant with a "normal" feed from division 12 and a "reserve feed from div.11. RCP trip circuits have a similar control power arrangement as this distractor).
Choice C is correct, see explanation above.
Choice D is incorrect, this would assume that circuit #1 is fed from bus 112 and circuit #2 is fed from bus 114 (misconception that there is both ESF and NON-ESF functions fed by the circuits. This is plausible since 1BwOA ELEC-3 allows the DG to be started with circuit #2 de-energized).

28 **ID: 1268031** **Points: 1.00**

Given:

- A release of 0WX01T, Liquid RadWaste Release Tank, is in progress with 0PR01J, LIQUID RELEASE EFFLUENT RAD MON, in service.

The following occurs:

- An audible alarm occurs on the RM-11.
- The 0PS101, LIQ RADWASTE, icon indicates DARK BLUE.
- The channel status display indicates a loss of sample flow has occurred.

With the above conditions, the 0PR01J is in ____(1)_____ failure and the release ___(2)___ AUTOMATICALLY isolate.

A. (1) EQUIPMENT
(2) did NOT

B. (1) EQUIPMENT
(2) DID

C. (1) OPERATE
(2) did NOT

D. (1) OPERATE
(2) DID

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #28
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: 073 Process Radiation Monitoring
K3.01 Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases
RO 3.6 SRO 4.2

Question meets K/A - question requires examinee knowledge of the effect of a PRM system malfunction on an effluent release.

TIER:	2
GROUP:	1
Task No:	R-AR-002
Obj No:	S.AR1-04-B-1

10CFR55 Link: 10CFR55.41(b)(11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Technical Reference with Revision Number: Big Note AR-1rev 10

Answer Explanation: A loss of sample flow will cause an operate failure and will terminate the release via the rad skid interlocks of closing the release tank pump discharge isolation valves. The distractor of an equipment failure would occur on a loss of process flow (vs. sample flow) and an equipment failure does not actuate the interlocks of the rad skid.

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

29 **ID: 1268032** **Points: 1.00**

Given:

- Unit 1 is at 100% power.
- 2SX005, Unit 0 CC Heat Exchanger Inlet Valve, is de-energized and closed for valve operator replacement.

Subsequently:

- 1A SX pump trips on overcurrent.
- 1B SX pump can NOT be started.
- Unit 1 reactor is tripped.
- A feedwater isolation occurs.
- Both Unit 1 AF pumps have automatically started.

Given the conditions above and assuming SX can NOT be restored to Unit 1, within the next 30 minutes, which of the following is correct?

A. 1A AF pump MUST be shutdown AND 1B AF pump may continue to operate.

B. 1B AF pump MUST be shutdown AND 1A AF pump may continue to operate.

C. BOTH Unit 1 AF pumps MUST be shutdown.

D. BOTH Unit 1 AF pumps may continue to operate.

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: #29
History: Bank from 2014 NRC exam
RO level
High Cog

K/A: 076 Service Water System
K3.07Knowledge of the effect that a loss or malfunction of the SWS will have on the following: ESF loads
RO 3.7 SRO 3.9

Meets K/A, examinee must know which AF pump can continue to operate on loss of SX based upon pump design.

TIER:	2
GROUP:	1
Task No:	R-AF-008
Obj No:	S.AF1-04

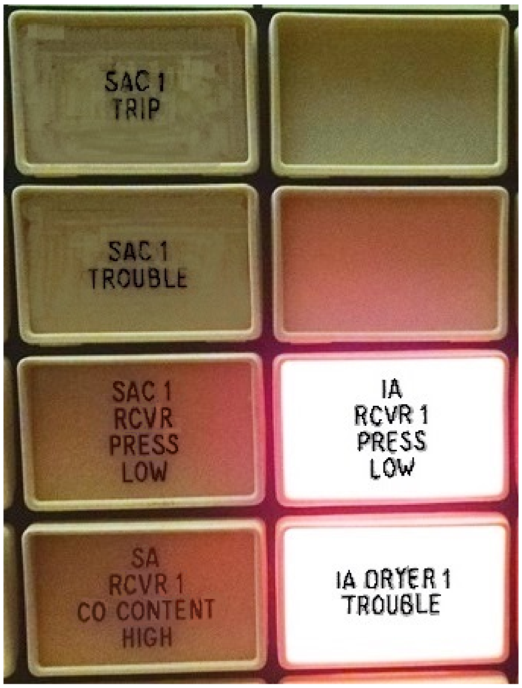
10CFR55 Link: 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Technical Reference with Revision Number: Lesson plan I1-AF-XL-01rev. 5e, page 12, big note AF-1 rev 21
Answer Explanation:
With neither Unit 1 SX pump available and the SX cross tie valve closed, no SX flow is available to cool the 1A AF pump. The 1B AF pump can run without a SX pump running because it has its own shaft driven SX booster pump. Per IER L1-11-4 Braidwood response for a loss of all AC associated with the operation of the Diesel Driven AF pump (DDAF), worst case scenario the 1B AF pump can run for 1 hour (conservative estimate) before the potential exists for pump and engine overheating. With local operator action, that time can be extended indefinitely by valving in the FLEX SX supply bypass line to the 1B AF pump.

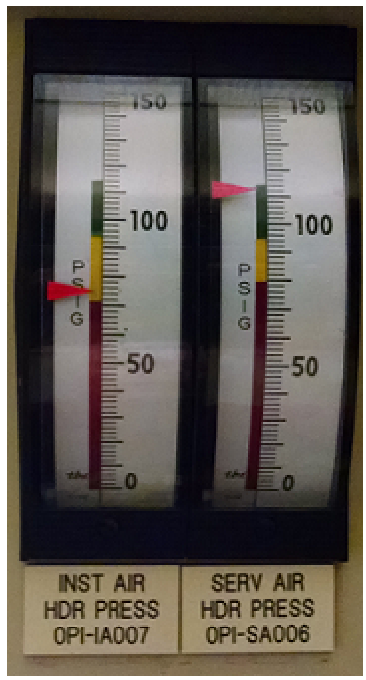
A is correct. See explanation.
B is incorrect. See explanation.
C is incorrect. See explanation.
D is incorrect. See explanation.

30 | **ID: 1271448** | **Points: 1.00**

Given:

- Unit 1 is at 100% power, normal alignment.
- The U-1 Station Air Compressor is in operation.
- The U-1 Instrument Air Dryer is aligned.
- Annunciators alarm and the RO notes the following 2 alarms LIT on 0PM01J.
- NO other alarms are LIT.
- Service air pressure has been stable as indicated below.
- Instrument Air pressure has trended down from 108 psig to its current value below.





As a result of the above conditions, the automatic plant response is...

A. a standby station air compressor STARTED.

B. the U-1 instrument air dryer bypass valve OPENED.

C. the U-1 station air compressor unloader valve went full CLOSED.

D. BOTH U-1 instrument air dryer towers simultaneously went to drying mode for 5 minutes.

Answer: B

Answer Explanation
Bwd 2016 NRC Exam Question: #30 **History:** New for 2016 Bwd NRC exam RO level High Cog **K/A:** 078 Instrument Air System (IAS) A3.01 Ability to monitor automatic operation of the IAS, including: Air pressure RO 3.1 SRO 3.2 The question meets the K/A, requires examinee ability to monitor an automatic operation of the instrument air system. TIER: 2 GROUP: 1 Task No: R-IA-002 Obj No: 3C.IA-01-A **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** BwAR 0-38-D3 rev. 7 **Answer Explanation:** The indications above (normal station air pressure with low instrument air pressure are representative of a failing (plugging) instrument air dryer. A drop in Instrument Air header pressure to <80 psig will cause the dryer bypass valves to auto open. Choice A is incorrect, but plausible because a low air pressure condition in the station air receiver will cause the standby compressor to start, however the station air header pressure is unaffected. Choice B is correct, see explanation above. Choice C is incorrect, but plausible because the station air compressor unloader valve normally throttles closed to maintain station air compressor discharge pressure. However, the condition described has normal station air pressure and the unloader would not have to change position. Choice D is incorrect, but plausible because the instrument air drying towers (2 per dryer) alternate between a 5 minute drying cycle and a 5 minute regenerating cycle every 5 minutes. However, they operate on a timer instead of responding to instrument air demand.

31	ID: 1268048	Points: 1.00

Given:

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

Subsequently the following occurs:

- A loss of bus 142 followed by a U-1 SI and Phase A isolation.

With the above conditions, the status of the RCFC WO containment penetrations will be...
(consider ISOLATED to mean WO water prevented from flowing OUT of containment from the inlet or outlet header penetrations)

A. NEITHER train of penetrations ISOLATED by either a closed MOV or a check valve.

B. 0A train penetration ISOLATED by either a closed MOV or a check valve AND 0B train NOT isolated.

C. 0B train penetration ISOLATED by either a closed MOV or a check valve AND 0A train NOT isolated.

D. BOTH trains of penetrations ISOLATED by either a closed MOV or a check valve.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #31
History: Bank from Bwd Systems
RO level
High Cog

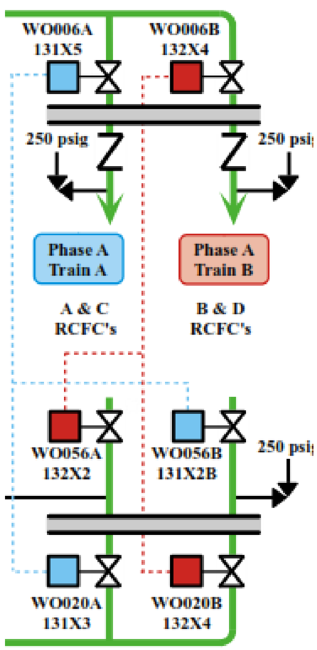
K/A: 103 Containment System
A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation
RO 3.5 SRO 3.8
The question meets the K/A because the candidate must predict the impacts of a containment phase A. 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.

TIER: 2
GROUP: 1
Task No: R-EP-037
Obj No: 3D.EP-19-A

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Technical Reference with Revision Number: Big Note WO-2 rev. 5.

Answer Explanation: Containment isolation WO valves are arranged in the below



configuration.
Each train of containment penetrations has one isolation valve that is powered from the opposite divisions electrical bus. This configuration prevents a containment penetration from not isolating with a loss of a single train of ESF power. Therefore, both penetrations will be isolated even though bus 142 (132X) lost power.
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.

32	ID: 1268215	Points: 1.00

Given:

- Unit 1 is at 100% power, normal alignment.
- The U-1 Boric Acid Storage Tank system is aligned and running on recirculation per BwOP AB-10, RECIRCING A BORIC ACID TANK.
- An EO performing field rounds reports the following to the MCR:
    - The U-1 Boric Acid Filter DP is at maximum (pegged high).
    - The U-1 Boric Acid system recirculation flow has dropped to ZERO.
    - 1AB03P, Boric Acid Transfer Pump, is running but NOT damaged.

- In response to the plugged filter, the U-1 RO then stops 1AB03P, Boric Acid Transfer Pump.

With the above conditions, and NO further operator actions, if Unit 1 requires blended flow make-up to the VCT, boric acid flow from 1AB03P...

A. IS available, and NO TRM entry is required.

B. is NOT available, and NO TRM entry is required.

C. is NOT available, and TRM 3.1.b, BORATION FLOWPATHS OPERATING, entry is required.

D. is NOT available, and TRM 3.1.f, BORATED WATER SOURCES OPERATING, entry is required.

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #32
History: New for Bwd 2016 NRC exam
RO level
High Cog

K/A: 004 Chemical and Volume Control System
K6.10 Knowledge of the effect of a loss or malfunction on the following CVCS components: Boric acid storage tank/boron injection tank recirculation flow path
RO 2.7 SRO 3.1
The question meets the K/A because the candidate must know the effects of a malfunction in the recirc flowpath of the AB system.

TIER:	2
GROUP:	1
Task No:	R-CV-005
Obj No:	S.CV2-07-A

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Technical Reference with Revision Number: Big note CV-4 rev 9

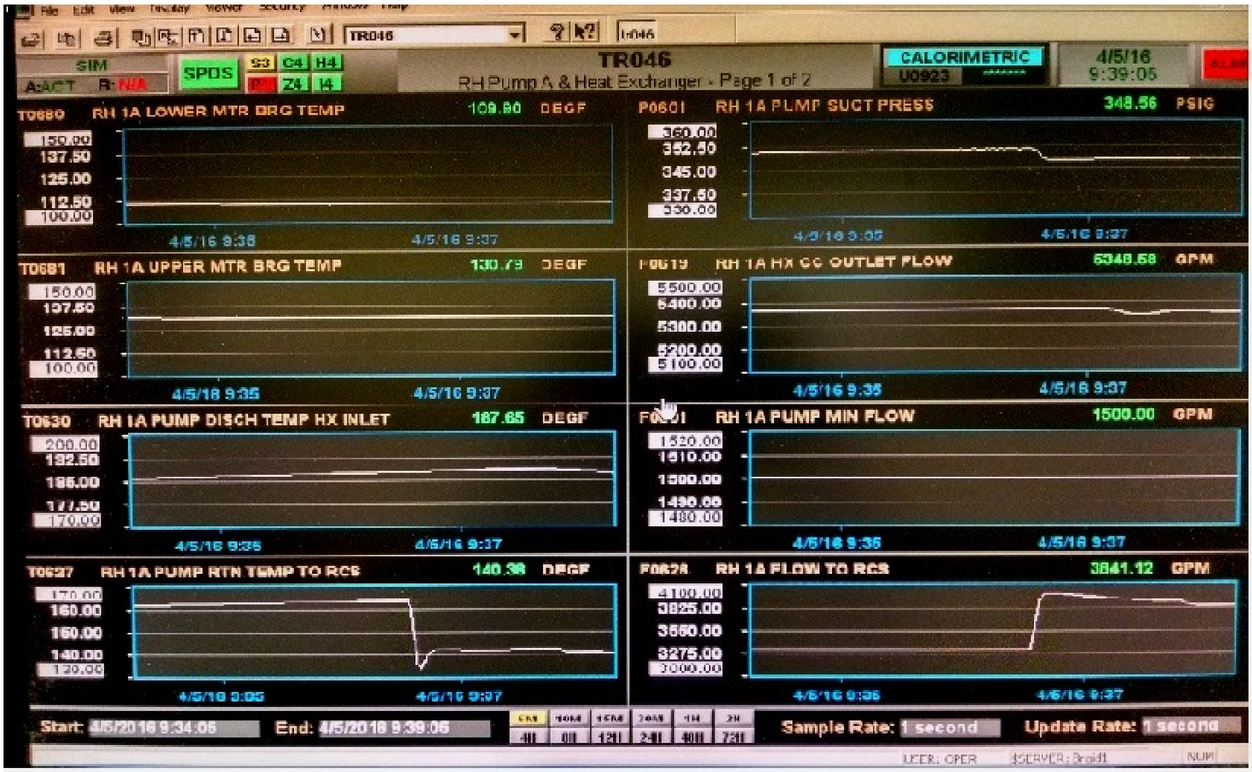
Answer Explanation: Normal standby alignment of a AB pump has a mini flow recirc path open that bypasses the AB filter. When a boric acid transfer pump is recirculating a boric acid storage tank (BAST) through the filter, the discharge is aligned back to the BAST and the filter bypass line is isolated. Flow is available to VCT make-up by manually or auto opening 1CV110A, however this flowpath is through the filter. If the filter becomes plugged while in recirc, there is no flowpath available to VCT make-up because the filter bypass line is isolated when aligned for recirc. Therefore, in the conditions given, no flow will be available to the VCT make-up line. TRM 3.1.b requires:
One boron injection flow path via the Chemical & Volume (CV) Control System from the Refueling Water Storage Tank (RWST) shall be OPERABLE, and either:
1. One additional OPERABLE flow path from the RWST, or
2. An OPERABLE flow path via a boric acid transfer pump from the Boric Acid Storage System

In normal plant alignment, 2 flowpaths are available from the RWST via 1CV112D and 1CV112E, therefore the loss of the flowpath from the BAST does not require a TRM 3.1.b entry. TRM 3.1.f is only required when TRM 3.1.b is also required, therefore TRM 3.1.f entry is also not required in normal plant configuration.
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.

33	ID: 1268226	Points: 1.00

Given:

- Unit 1 is in Mode 5 with the 1A RH train in shutdown cooling mode.
- The RO was monitoring RH system performance on the plant process computer (PPC) when the following trends were noted (see below PPC screen shot).



With the above conditions, which of the following equipment malfunctions caused the noted indications?

A. Instrument Air was lost to 1RH606, HX 1A FLOW CONT VLV.

B. Inadvertent stroke of 1RH610, PP 1A MINIFLOW VLV.

C. Instrument Air was lost to 1RH618, HX 1A BYP FLOW CONT VLV.

D. Inadvertent stroke of 1CC9412A, CC TO RH HX 1A ISOL VLV.

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: #33
History: New for Bwd 2016 NRC exam
RO level
High Cog

K/A: 005 RHRS
Gen 2.1.19 Ability to use plant computers to evaluate system or component status
RO 3.9 SRO 3.8

The question meets the K/A because the candidate must be able to evaluate system status based upon the PPC.

TIER:	2
GROUP:	1
Task No:	R-RH-004
Obj No:	4C.RH-05

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Technical Reference with Revision Number: ILT lesson plan I1-RH-XL-01 rev. 5d page 29, big note RH-1b rev 0

Answer Explanation: The significant trend changes are RH flow (rose approx. 500 gpm) and RH return temperature to RCS (dropped approx. 30°F). These indications represent a failure of the 1RH606 (open) which will raise flow through the RH HX (and total system flow).
Choice A is correct, see explanation above.
Choice B is incorrect, a stroke of 1RH610 would temporarily lower total system flow until 1RH618 opens to compensate and raise system flow back to original value of 3300 gpm.
Choice C is incorrect, failing air to 1RH618 would cause the valve to fail closed and system flow to drop.
Choice D is incorrect, 1CC9412A stroking close would raise return temperature (loss of cooling) but not significantly change system flow. The subtle change in CC flow on the PPC represented a change in the water density when the CC temperature heated up (higher heat transfer rate).

34	ID: 1268235	Points: 1.00

Given the following sequence of events:

- Unit 1 was at full power, normal alignment.
- A U-1 SI actuation then occurs.

With the above conditions, which of the following Group 2 status lights would be the LAST to illuminate? (assume all valves have approximately the same stroke times)

A. 1CV112B/C CLOSED

B. 1CV112D/E OPEN

C. 1SI8801A/B OPEN

D. 1SI8105/8106 CLOSED

Answer: A

Answer Explanation
Bwd 2016 NRC Exam Question: #34 **History:** New for Bwd 2016 NRC exam RO level High Cog **K/A:** 006 Emergency Core Cooling KA4.05 Ability to manually operate and/or monitor in the control room: Transfer of ECCS flowpaths prior to recirculation RO 3.9 SRO 3.8 The question meets the K/A because the candidate must know interlocks of ECCS valves that position prior to recirc phase and predict how the indications will appear in the MCR. TIER: 2 GROUP: 1 Task No: R-EF-001 Obj No: S.EC1-09-B **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features **Technical Reference with Revision Number:** 20E-1-4030CV10 rev O. **Answer Explanation:** Two pieces of information are necessary to predict which valve lights will light last. One is whether the valve status lights change status when the valve begins to stroke or at end of stroke. Although, some of these valves have multiple status light indications that are actuated by different limit switches for the same valve, all of the ECCS group 2 valves actuate with their respective limit switch for the valve emergency position. Therefore, the only reason one set of valves would be delayed in operating after the SI signal is received would be an interlock. In this case 1CV112B/C valves are interlocked to wait to close until 1CV112D/E valves are open (train specific) to ensure CV pump suction is never lost during the transition from normal charging to ECCS injection phase. 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.

35 **ID: 1268231** **Points: 1.00**

Given:

- Unit 1 is at full power, normal alignment.

An event then occurs and the following indications are noted:

- Annunciator 1-2-E4, CC SURGE TANK AUTO-M/U ON, alarms and the following SER points print:
  - -1434 VALVE 1CC183 NOT CLOSED
  - -1430 VALVE 1CC182 NOT CLOSED

- Annunciator 1-2-A5, CC SURGE TANK LEVEL HIGH/LOW
  - -0417 CC SURGE TANK LEVEL (PUMP 1A SUCT) LOW
  - -2147 CC SURGE TANK LEVEL (PUMP 1B SUCT) LOW

- Annunciator 1-2-E3, SX TO CC SURGE TANK TROUBLE, alarms and the following SER points print:
  - -0266 SX M/U TO CC SURGE TK VALVE 1CC202A OPEN
  - -0264 SX M/U TO CC SURGE TK VALVE 1CC201A OPEN

- NO other annunciators NOR SER points related to the CC system have alarmed OR cleared.
- The RO notes CC Surge Tank level is currently STABLE.

With the above conditions, which of the following is the currently expected CC surge tank level range?

A. 0% - 22%

B. 23% - 30%

C. 31% - 45%

D. 46% - 50%

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #35
History: New for 2016 NRC exam
RO level
High Cog

K/A: 008 Component Cooling Water System (CCWS)
A3.01 Ability to monitor automatic operation of the CCWS, including: Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS
RO 3.2 SRO 3.0

The question meets the K/A because examinee must know the CC system setpoints for CC surge tank make-up auto operations and evaluate the corresponding alarms to those setpoints.

TIER:	2
GROUP:	1
Task No:	R-CC-003
Obj No:	S.CC1-10

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Technical Reference with Revision Number: BwAR 1-2-E3 rev 01

Answer Explanation: CC surge tank make up occurs at the following setpoints:
50% - 1CC183, WM supply to CC surge tank opens
45% - 1CC182, PW supply to CC surge tank opens
30% - 1CC201A and 1CC202A, SX supply to CC surge tank (A train) open.
22% - 1CC201B and 1CC202B, SX supply to CC surge tank (B train) open.

A is incorrect. In this range the B train of SX make-up would also open. Because the sx is a double isolation and other sources are single isolation, the two SX supply valves being open makes it plaussible that the valves are for two seperate SX supplies.
B is correct. See explanation.
C is incorrect. This would be the range if the WM and PW make-up were only open (not SX).
D is incorrect. This would be the range if the WM make-up was only open (not SX nor PW).

36	ID: 1268271	Points: 1.00

Given:

- Unit 1 has had a reactor trip and SI from full power.
- The crew is currently performing 1BwEP-0, REACTOR TRIP OR SI, step 7.a. Group 2 RCFC Accident Mode status lights - LIT
- The RO notes that ONE of the FOUR RCFC Accident Mode status lights is NOT LIT.

With the above conditions, the single component that failed to realign to accident configuration is a...

A. Reactor Containment Fan Cooler.

B. SX Containment Isolation Valve.

C. SX Containment Chiller Isolation Valve.

D. SX Containment Chiller Bypass Valve.

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: #36
History: New for Bwd 2016 NRC exam
RO level
High Cog

K/A: 022 Containment Cooling System
Gen 01.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.
RO 4.3 SRO 4.4

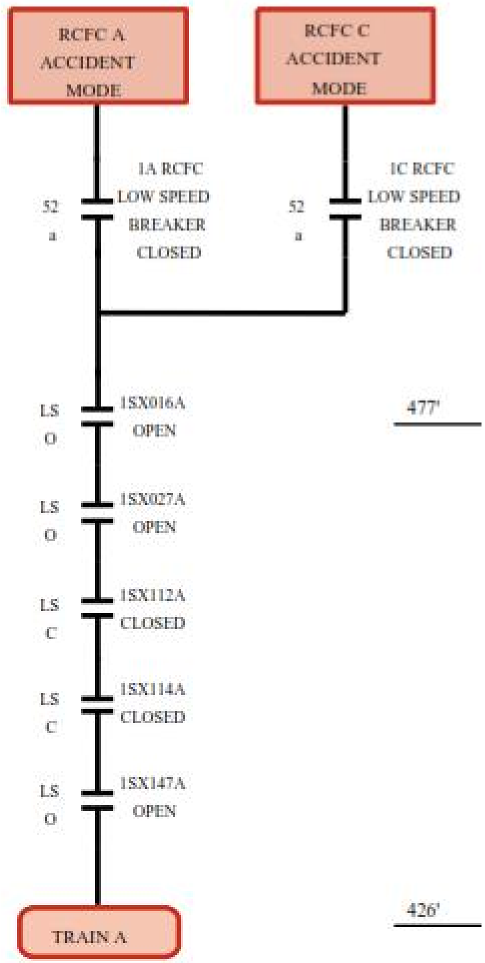
The question meets the K/A because the candidate must know which component to manually re-configure in the RNO column of 1BwEP-0.

TIER:	2
GROUP:	1
Task No:	R-VP-011
Obj No:	S.VP1-06

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Technical Reference with Revision Number: Big note VP-3 rev 6

Answer Explanation: The RCFC accident mode lights have inputs from 5 separate valves limit switches and the RCFC fan low speed breaker auxiliary contacts. All of the valve limit switches feed two individual lights, while the RCFC breaker contacts are specific to each light. Therefore, if only one light is not lit, the component that is specific to the individual light is the RCFC. See electrical contact diagram below.

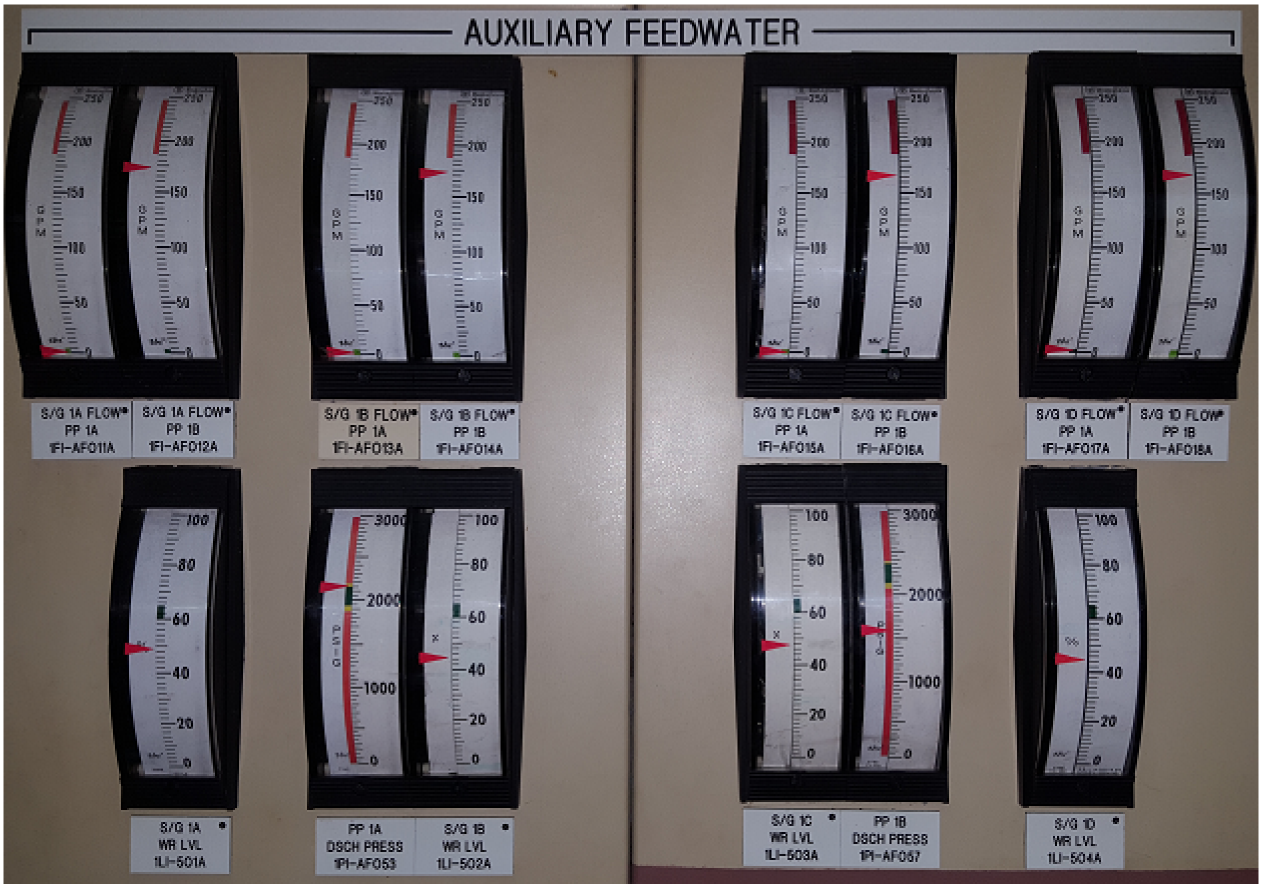


Choice A is correct, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

37	ID: 1268276	Points: 1.00

Given:

- Unit 1 was operating at full power when a near simultaneous loss of an electrical bus and SI occurred
- 5 minutes later, the RO notes the following indications on 1PM06J.
- NO operator actions have been taken since SI actuated.



With the above conditions, which of the following was the sequence of events that occurred?

A. AC Instrument Bus 111 was lost BEFORE the SI signal.

B. AC Instrument Bus 111 was lost AFTER the SI signal.

C. DC Bus 111 was lost BEFORE the SI signal.

D. DC Bus 111 was lost AFTER the SI signal.

Answer: B

Answer Explanation
Bwd 2016 NRC Exam Question: #37 **History:** New for Bwd 2016 NRC exam RO level High Cog **K/A:** 061 Auxiliary/Emergency Feedwater (MFW) System Gen 2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions. RO 4.2 SRO 4.4 The question meets the K/A, requires examinee ability to interpret control room indications to verify the status of the system. Question only tests first part of K/A IAW NUREG ES-401 section D.2.a (page 6) because trying to test both parts does not create a psychometrically sound question. TIER: 2 GROUP: 1 Task No: R-AF-002 Obj No: S.AF1-15 **10CFR55 Link:** 10CFR55.41(b)(4) Secondary coolant and auxiliary systems that affect the facility. **Technical Reference with Revision Number:** Big note AF-1 rev 21. **Answer Explanation:** A loss of Instrument Bus 111 after the SI signal would cause the 1AF005A-D valves to fail close (loss of power to controllers) and flow to drop to zero in the 1A AF train. Choice A is incorrect, if Instrument Bus 111 failed before the SI, the 1A AF pump would not have started and zero 1A discharge pressure would be indicated. Choice B is correct, see explanation above. Choice C is incorrect, if DC Bus 111 failed before the SI, the 1A AF pump would not have started and zero 1A discharge pressure would be indicated. Choice D is incorrect, if DC Bus 111 failed after the SI, the 1A AF pump would have started and delivered flow (1AF005 valves would be properly throttled.

38	ID: 1268284	Points: 1.00

Given:

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

A MCR annunciator alarms on 1PM01J and the RO notes the following indications:

Currently, with above conditions, which of the following malfunctions occurred?

A. DC Battery Charger 112 Supply Breaker TRIPPED.

B. DC Battery Charger 112 Output Breaker TRIPPED.

C. DC Bus 114 supply fuses OPENED.

D. DC Bus 112 Main Feed Breaker TRIPPED.

Answer: C

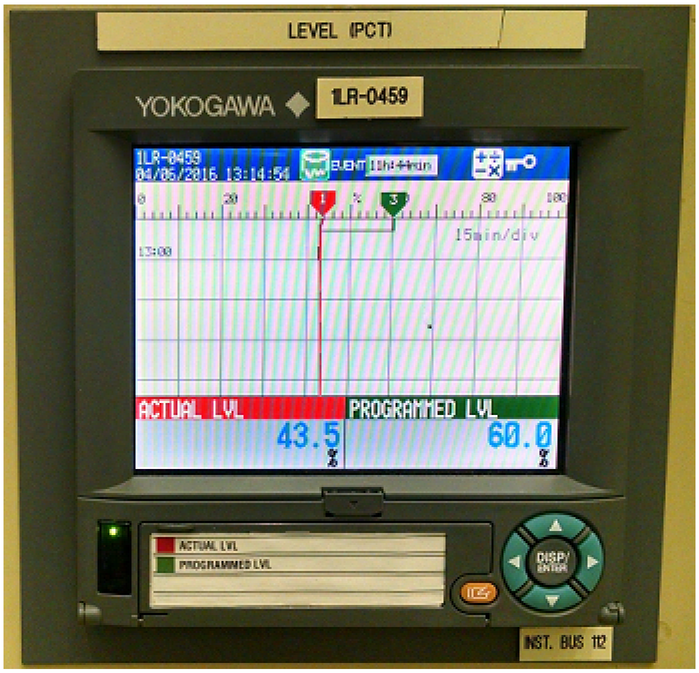
Answer Explanation
Bwd 2016 NRC Exam Question: #38 **History:** New for 2016 Bwd NRC exam RO level High Cog **K/A:** 063 D.C. Electrical Distribution A4.01 Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses. RO 2.8 SRO 3.1 The question meets the K/A by requiring the ability to determine status of DC bus supply breakers from MCR indications. TIER: 2 GROUP: 1 Task No: R-DC-003 Obj No: S.DC1-09 **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** Big note DC-1 rev 9. **Answer Explanation:** The combination of annunciator 1-22-E10 lit (which receives inputs from both DC bus 112 and 114) and DC bus 112 voltage meter reading normal (in green band) indicates a loss of DC bus 114. DC bus 114 is fed by supply fuses from DC bus 112. Choice A is incorrect, DC Battery Charger 112 Supply Breaker TRIPPED would cause alarm 1-22-E7 and a much lower DC bus voltage indication. Choice B is incorrect, DC Battery Charger 112 Output Breaker TRIPPED would cause alarm 1-22-E8 and a much lower DC bus voltage indication. Choice C is correct, see explanation above. Choice D is incorrect, DC Bus 112 Main Feed Breaker TRIPPED would cause DC bus 112 voltage meter to read 0 volts.

39 **ID: 1268328** **Points: 1.00**

Given:

- Unit 1 is at 50% power.
- Control Rods are in MANUAL.
- All other equipment is in a normal alignment.

An event then occurs and the RO notes the below indications on 1LR-0459, PZR Level Recorder.



Which of the following failed HIGH would cause the above indications?

A. A power range NI detector

B. A turbine impulse pressure transmitter

C. A narrow range Tcold RTD

D. An RCS loop Delta T channel

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #39 **History:** New for 2016 Bwd NRC exam RO level High Cog **K/A:** APE 028 Pressurizer Level Malfunction A2.01 Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: pressurizer level indicators and alarms RO 3.4 SRO 3.6 The question meets the K/A by requiring the ability to monitor pressure control instrumentation and determine failed input. TIER: 1 GROUP: 2 Task No: R-RY-015 Obj No: S.RY1-20 **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** Big note RY-3 rev 11. **Answer Explanation:** The sudden change in program pressurizer level indicates a failure of an input. Program pressurizer level is controlled by auctioneered high Tave. Choice A is incorrect, but plausible because PRNIs feed other control and trip circuits including OTDT, rod control and turbine feedback loops. Choice B is incorrect, but plausible because turbine impulse pressure feeds the Tref circuit. Choice C is correct, see explanation above. Choice D is incorrect, but plausible because delta Ts feed other control and trip circuits including OTDT, OPDT and control rod insertion limits.

40	ID: 1268342	Points: 1.00

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

A. 1AC

B. 2AC

C. 1BD

D. 2BD

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: # 40
History: Bank from 2011 Bwd NRC Exam
RO level
High Cog

K/A: APE 003 Dropped Control Rod
Knowledge of the interrelations between the Dropped Control Rod and the following:
AK2.05 Control rod drive power supplies and logic circuits
RO 2.5 SRO 2.8

The question meets the K/A, requires examinee knowledge of the interrelations between a dropped rod and the logic circuits in the rod control system.

TIER: 1
GROUP: 2
Task No: R-OA-065
Obj No: 3D.OA-75
10CFR55 Link: 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Technical Reference with Revision Number: Lesson plan I1-OA-XL-34 rev 11, page 10., 1BwOA ROD-3 rev 106 page 20

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

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.

41 **ID: 1268388** **Points: 1.00**

Given:

- A release of the 0A Gas Decay Tank (GDT) is in progress per BwOP GW-500T1, GAS DECAY TANK RELEASE FORM.
- Subsequently, the 0B (on line) GDT relief valve fails (sticks) OPEN.
- The rise in flow past the 0PR02J, GDT EFFLUENT, rad monitor causes the skid to exceed the high radiation ALARM setpoint.

With the above conditions...

A. BOTH the 0A and 0B GDT tanks effluent will be AUTOMATICALLY isolated.

B. the 0A GDT tank effluent must be MANUALLY isolated, but the 0B GDT effluent can NOT be isolated.

C. BOTH the 0A and 0B GDT tanks effluent will be MANUALLY isolated by CLOSING 0GW014, WASTE GAS DISCH VLV.

D. the 0A GDT tank effluent will be AUTOMATICALLY isolated, but the 0B GDT effluent can NOT be isolated.

Answer: D

Answer Explanation
Bwd 2016 NRC Exam Question: #41 **History:** Bank from 2011 NRC exam RO level High Cog **K/A:** APE 060 Accidental Gaseous Radwaste Release Ability to monitor automatic operation of the Waste Gas Disposal System including: GEN 04.31 Knowledge of annunciator alarms, indications, or response procedures. RO 4.2 SRO 4.1 The question meets the K/A, requires examinee knowledge of waste gas system rad monitor high alarm. TIER: 1 GROUP: 2 Task No: R-GW-001 Obj No: 3S.GW-01 **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** Big note RW-1 rev 4 **Answer Explanation:** The waste gas system is configured such that the tank relief valves discharge header intersects the waste gas release header upstream of the 0PR02J rad monitor sample line, so the rad monitor will detect the rise in flow and subsequent rise in rad levels when a relief valve opens. However, the waste gas release FCV, 0GW014, is upstream of the relief header intersection point. Therefore, a high rad signal, which automatically closes the 0GW014, will isolate the tank that has a release in progress, but will not isolate a tank with a failed relief valve. Additionally, no isolation exists for the relief valve header, so it can not be manually isolated. Choice A is incorrect, 0B tank can not be isolated. Choice B is incorrect, 0A tank will be automatically isolated. Choice C is incorrect, 0A tank will be automatically isolated, 0B tank can not be isolated. Choice D is correct, see explanation above.

42 **ID: 1268389** **Points: 1.00**

Given:

- A fire has occurred in the plant and annunciator 0-37-A4, UNIT 1 AREA FIRE, is in alarm.
- The RO is reviewing BwAR 0-37-A4 for applicable actions.
- It is determined that the fire meets criteria (in applicable fire zone and of sufficient magnitude) for specific actions listed in BwAR 0-37-A4.

With the above conditions, which of the following describes the actions listed in BwAR 0-37-A4?

A. Start the Auxiliary Feedwater Pumps to prevent a loss of an adequate heat sink.

B. Start the Emergency Diesel Generators and run unloaded to ensure readiness for a loss of an ESF bus.

C. Transfer the non-ESF bus feeds to the SATs and remove UAT feed breaker control power fuses to prevent a failure of an auto bus transfer from affecting on site power.

D. Swap the U-1 CV pump suction from the VCT to the RWST and de-energize the suction valves to prevent a loss of operating CV pump suction.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #42
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: APE 067 Plant Fire On Site
K3.02 Knowledge of the reasons for the following responses as they apply to the Plant Fire on Site: Steps called out in the site fire protection plan, FPS manual, and fire zone manual.
RO 2.5 SRO 3.3

The question meets the K/A, requires examinee knowledge of steps provided in response to plant fire procedure.

TIER:	1
GROUP:	2
Task No:	R-FP-002
Obj No:	S.FP1-18

10CFR55 Link: 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility
Technical Reference with Revision Number: BwAR 0-37-A4 rev 14

Answer Explanation: A note in BwAR 0-37-A4 reads A fire in any of the following Detection Zones may cause an inadvertent closure of 1CV112B or 1CV112C resulting in loss of suction to a running CV pump:
(lists multiple fire zones). Further immediate operator actions state: IF the fire is of sufficient magnitude to damage plant equipment in any of the Zones noted above, THEN PERFORM the following as a precautionary measure to prevent losing suction to the operating CV pump:
VERIFY/OPEN and REMOVE power from 1CV112D and/or 1CV112E.
VERIFY/CLOSE and REMOVE power from 1CV112C.

Choice A is incorrect, but plausible because starting the AF pumps pre-emptively is an action in 1BwOA SEC-1.
Choice B is incorrect, but plausible because starting the DGs in standby is a precautionary action taken in 1BwOA SECURITY-1.
Choice C is incorrect, but plausible because BwOP FP-100T35 (attachment for fire in Aux. Elec. Equip Room) contains steps for removing control power for SAT feed breakers to prevent them from inadvertently closing in on a bus being supplied by a DG.
Choice D is correct, see explanation above.

43	ID: 1268412	Points: 1.00

Given:

- Unit 1 was at 100% power, normal alignment.

- An event occurred causing a Phase A containment isolation.
- Five minutes after the Phase A actuated, the RO noted that several of the Phase A isolation valves did NOT close automatically.
- After the event recovery, the crew is investigating which train of Phase A did NOT auto actuate.
- The US suspects that a SSPS master relay failed.

Which of the following pairs of valves will the operators use to determine which train of Phase A failed to actuate?
(assume only one valve in each pair automatically closed)

A. 1CV8100/8112, SEAL WTR RTRN CNMT ISOL VLVS

B. 1CV8149A/B, LTDWN ORIF ISOL VLVS

C. 1CC9414/9416, CC FROM RC PUMPS ISOL VLVS

D. 1SD005A/B, S/G 1A/1D BLWDN SAMPLE ISOL VLVS

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: # 43
History: Modified from Bwd 2011 NRC Exam (RS10103-N01)
RO level
Low Cog

K/A: APE 069 Loss of Containment Integrity
A1.01 Ability to operate and / or monitor the following as they apply to the Loss of Containment Integrity: Isolation valves, dampers, and electropneumatic devices
RO 3.5 SRO 3.7

The question meets the K/A, requires examinee ability monitor isolation valve failures to determine the cause.

TIER:	1
GROUP:	2
Task No:	S-OA-067
Obj No:	T.OA23-07

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Technical Reference with Revision Number: 1BwOA PRI-13 rev 100 page 2

Answer Explanation:
Choice A is correct, 1CV8100/8112 are motor operated valves that are operated by separate train relay actuations. Additionally, in the event of an inadvertent Phase A, 1BwOA PRI-13 specifically uses these valves to determine which train actuated.
Choice B is incorrect, although 1CV8149A/B receives individual train containment isolation signal (CIS), because the valves are inside containment, either train will isolate instrument air to containment and the valve will fail closed. Therefore it is not positive indication of a single train actuation.
Choice C is incorrect, 1CC9414/9416 receive a Phase B (NOT Phase A) CIS. Therefore it is not positive indication of a phase A actuation.
Choice D is incorrect, 1SD005A/B are single valve isolation of containment penetrations that receive a CIS signal from both trains.

44 **ID: 1269473** **Points: 1.00**

Given:

- Unit 1 was at 100% power, normal alignment.

Then the following occurred:

- An RCS LOCA.
- The reactor tripped, coincident with a loss of offsite power (LOOP).
- Bus 141 is ENERGIZED.
- Bus 142 is FAULTED.
- The 1A CV pump TRIPPED.
- The inability to obtain adequate ECCS injection flow has caused an Inadequate Core Cooling condition.
- The crew is attempting to depressurize the SGs per 1BwFR-C.1, RESPONSE TO INADEQUATE CORE COOLING.

With the above conditions, INITIALLY the ONLY available SG PORVs that can be used to MANUALLY depressurize the SGs from the MCR are the 1MS018...

A. A & D

B. C & D

C. A, B & D

D. A, C & D

Answer: D

Answer Explanation
Bwd 2016 NRC Exam Question: # 44 **History:** New for Bwd 2016 NRC exam RO level High Cog **K/A:** APE 074 Inadequate Core Cooling Gen 02.37 Ability to determine operability and/or availability of safety related equipment RO 3.6 SRO 4.6 The question meets the K/A, requires examinee ability to determine which SG PORVs will be available for the SG depressurization. TIER: 1 GROUP: 2 Task No: R-FR-019 Obj No: S.MS1-19 **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** Big note MS-7 rev 8 MS PORV CONTROLS **Answer Explanation:** The SG PORV pump and control power are fed from 480v ESF busses. PORV 1A and 1D are fed from bus 131X. PORV 1B and 1C are fed from bus 132X. With bus 142 de-energized, bus 132X will be de-energized. Therefore the 1B and 1C PORVs will not have their normal power supply. However, the 1C and 1D PORVs have a UPS (battery) backup power supply (a recent plant modification), so the 1C PORV will be available along with 1A and 1D which still have their normal power supply from bus 131X via bus 141. Choice A is incorrect, but plausible because A and D PORVs would be correct before the UPS modification. Choice B is incorrect, but plausible because C and D are the PORVs with UPS. Choice C is incorrect, but plausible if the candidate confuses the normal power supplies and believes A and B are powered from division !1 ESF power. Choice D is correct, see explanation above.

45	ID: 1268468	Points: 1.00

Given:

- An event has occurred on Unit 1 causing elevated containment radiation levels.
- The crew has entered emergency procedures and is performing mitigation actions in the procedures.
- The US assigned the BOP the task of monitoring the RM-11 for "Adverse Containment" conditions.

The BOP will check for containment radiation levels on rad monitor...

A. 1PR01J, CNMT PURGE.

B. 1AR011J, CONTAINMENT FH ICDT.

C. 1PR11J, CNMT ATMOS.

D. 1AR020J, CONTAINMENT HI RNG.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #45
History: Bank from Bwd EOPS
RO level
Low Cog

K/A: EPE W/E16 High Containment Radiation
A2.02 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments

The question meets the K/A, requires examinee knowledge of adverse containment condition criteria which is then used in decision making process of emergency procedure flowpaths.

TIER:	1
GROUP:	2
Task No:	R-EP-009
Obj No:	T.EP00-06

10CFR55 Link: 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility

Technical Reference with Revision Number: 1BwEP-0 rev. 207, page 3.

Answer Explanation: The Braidwood emergency procedures have multiple steps that require different parameter criteria depending upon containment status (adverse vs. not adverse). A note at the beginning of 1BwEP-0 defines the criteria for adverse containment and is applicable throughout the WOG EP series. The note lists 1AR020/021 rad skids as the rad monitor to determine adverse containment during high rad conditions. The distractors are all other containment radiation monitors used in either emergency plan procedures or abnormal operating procedures.
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.

46 | **ID: 1268474** | **Points: 1.00**

Given the following sequence of events:

- Unit 1 has experienced a small break RCS LOCA.
- RCS pressure is currently 2200 psig and stable.
- The crew is performing 1BwEP ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, and is about to initiate an RCS cooldown using steam dumps.

As PZR pressure drops during the cooldown, which of the following is the MAXIMUM indicated PZR pressure that the NSO can block the Steamline Low Pressure SI signal?

A. 2100 psig

B. 2000 psig

C. 1900 psig

D. 1800 psig

Answer: C

Answer Explanation

Bwd 2016 NRC Exam Question: #46
History: Bank from 2011 Bwd NRC Exam
RO level
Low Cog

K/A: W/E03 LOCA Cooldown and Depressurization
K1.03 Knowledge of the operational implications of the following concepts as they apply to the (LOCA Cooldown and Depressurization)
Annunciators and conditions indicating signals, and remedial actions associated with the (LOCA Cooldown and Depressurization)

The question meets the K/A, requires examinee indicating signals and remedial actions during a post LOCA Cooldown and Depressurization.

TIER:	1
GROUP:	2
Task No:	R-EP-003
Obj No:	T.EP02-01-D

10CFR55 Link: Cross Ref: 10CFR55.41(b) (7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Technical Reference with Revision Number: 1BwEP ES-1.2 rev 203 page 10.

Answer Explanation: 1BwEP ES-1.2, step 8.c has the crew check PZR pressure <1930 psig prior to blocking Low Steamline Pressure SI to prevent the MSIVs from closing. This cannot be performed until below the P-11 setpoint (1930 psig).

Choice A is incorrect, pressure is not yet below P-11.
Choice B is incorrect, pressure is not yet below P-11.
Choice C is correct, see explanation above.
Choice D is incorrect, pressure is below P-11, but stem asks for earliest pressure that SI can be blocked. It is important not to wait too long to block SI as the steam generators will be depressurizing also as the cooldown continues and the SI signal must be blocked before the setpoint is reached.

47 **ID: 1268475** **Points: 1.00**

Given the following sequence of events:

- Unit 1 has experienced an event resulting in natural circulation conditions.
- The crew is currently performing 1BwEP ES-0.2, NATURAL CIRCULATION COOLDOWN.

Per 1BwEP ES-0.2 Continuous Action Summary page, the crew will verify all ______(1)_______ are running. The bases for this step is to aid in cooling of the ______(2)_______.

A. (1) CRDM Fans
(2) reactor upper internal instrumentation

B. (1) CRDM Fans
(2) upper reactor vessel head

C. (1) Rx Cavity Vent Fans
(2) Source Range NIs

D. (1) Rx Cavity Vent Fans
(2) lower reactor vessel

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #47
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: W/E09 Natural Circulation Operations
K2.02 Knowledge of the interrelations between the (Natural Circulation Operations) 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

The question meets the K/A, requires examinee knowledge of alternate heat removal mechanism for the upper vessel head during natural circulation .

TIER:	1
GROUP:	2
Task No:	R-EP-002
Obj No:	T.EP01-06-D

10CFR55 Link: Cross Ref: 10CFR55.41(b) (10) Administrative, normal, abnormal, and emergency operating procedures for the facility

Technical Reference with Revision Number: BD-EP ES-0.2 rev 205, page 12 1BwEP ES-0.2 rev 203 page 3

Answer Explanation: 1BwEP ES-0.2, step 2 and CAS page, all available CRDM fans are to be verified running. This is to provide supplemental reactor upper head cooling to prevent a bubble from forming in the vessel head. Under natural circulation conditions, reactor vessel head bypass flow is minimal due to flow restrictions and as the RCS is cooled down and depressurized, a localized loss of subcooling could occur in the head.

Choice A is incorrect, but plausible because this is the CRDM fans normal function.
Choice B is correct, see explanation above.
Choice C is incorrect, but plausible because this is the Rx cavity vent fans normal function.
Choice D is incorrect, but plausible because this is also a normal at power function of the Rx cavity vent fans.

48 **ID: 1268488** **Points: 1.00**

Given:

- Unit 1 was MANUALLY tripped from 100% power due to lowering SG NR levels.
- All equipment operated as designed.
- Following the manual reactor trip, the 1A SG NR level dropped below the LO-2 setpoint.
- The Unit 1 NSO has just depressed the following annunciator push buttons associated with 1PM05J in the following order:
  1) SILENCE
  2) ACKNOWLEDGE
  3) RESET

- NO other operator actions have been taken on the 1PM05J panel.
- 1A SG NR level continues to remain below the LO-2 setpoint.

With the above conditions, which one of the following describes the Reactor Trip Annunciator Box status of the following?

1-11-A1, MANUAL RX TRIP
1-11-A8, S/G 1A LEVEL LO-2 RX TRIP

A. 1-11-A1 is LIT SOLID (RED) and 1-11-A8 is SLOW FLASHING (WHITE).

B. 1-11-A1 is SLOW FLASHING (RED) and 1-11-A8 is NOT LIT (DARK).

C. 1-11-A1 is SLOW FLASHING (RED) and 1-11-A8 is LIT SOLID (WHITE).

D. 1-11-A1 is SLOW FLASHING (RED) and 1-11-A8 is SLOW FLASHING (WHITE).

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #48 **History:** Bank from Bwd ILT systems RO level High Cog **K/A:** EPE 007 Reactor Trip K2.03 Knowledge of the interrelations between a reactor trip and the following: Reactor trip status panel RO 3.5 SRO 3.6 The question meets the K/A because the candidate must know the status of the reactor trip first out panel when multiple reactor trip signals are generated. TIER: 1 GROUP: 1 Task No: R-AN-006 Obj No: S.AN1-03-A **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** Lesson plan I1-AN-XL-01 rev. 2b, pages 12-14, big note AN-1 rev 5 **Answer Explanation:** The reactor trip status panel contains a logic circuit that will light the first reactor trip signal annunciator red and then subsequent reactor trip signals will light in white bulbs. The manual Rx trip will not be active (switch spring returns to normal) so it will be slow flashing red. The reset pushbutton does not clear (reset) the reactor trip status block. Alarms must be reset with a key switch that is unique to the first out block of annunciators. Active trip signals will remain lit solid. Choice A is incorrect, this is the opposite flashing/not flashing configuration. Choice B is incorrect, active trip signals do not clear. Choice C is correct, see explanation above. Choice D is incorrect, manual trip will be slow flashing because active signal is no longer in.

49	ID: 1268490	Points: 1.00

Which of the following conditions is the OTDT reactor trip designed to protect against?

A. An uncontrolled rod withdrawal from start-up conditions.

B. A trip of all feedwater pumps from at power conditions.

C. A stuck open pressurizer safety valve from at power conditions.

D. A main steam line break from at power conditions.

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #49 **History:** New for Bwd 2016 NRC exam RO level Low Cog **K/A:** APE 008 Pressurizer Vapor Space Accident Gen 02.38 Knowledge of conditions and limitations in the facility license. RO 3.6 SRO 4.5 The question meets the K/A, requires examinee knowledge of limitations in the facility license, specifically the requirements for protection against DNBR (hot leg saturation) conditions. **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. TIER: 1 GROUP: 1 Task No: R-EP-032 Obj No: S.RC1-07 **Technical Reference with Revision Number:** Lesson Plan I1-RP-XL-02 rev 3 page 6 and 9, Big note EF-1 rev **Answer Explanation:** The DNB accident family includes any event resulting in a degradation of the parameters associated with DNB. (i.e. temperature, pressure, flow or power including power distribution). Events causing RCS low pressure are mitigated by either a direct low pressure reactor trip or by an OTDT reactor trip. Events raising RCS temperature are mitigated by an OTDT trip and operation of SG PORVs and safety valves. Events of lowering pressurizer pressure are terminated by OTDT trip and backed up by the low pressurizer pressure trip. This is the type of event that a stuck open pressurizer safety valve is (vapor space LOCA). Choice A is incorrect, Start-up accidents are protected against by the Nuclear Instrument high neutron flux trips of the Source and Intermediate Ranges and the low power high neutron flux trip of the Power Range. Choice B is incorrect, Loss of Heat Sink accidents are protected against by the Lo-2 SG level trip and the Turbine Trip/Reactor Trip. Choice C is correct, OTDT is designed to prevent exceeding the DNBR safety limit at ALL power conditions. The OTDT function bounds the raising of Tave to hot leg saturation conditions at all appreciable power levels. Additionally, in Mode 2, if the RCS temperature is raised too high, the Main Steam safety valves will lift causing some RCS loop DT, to assure the OTDT will "see" a slow adverse reactivity trend. Choice D is incorrect, a Main Steam Line Break (over power condition) would be protected against by OPDT and backed up by low steam line pressure SI/RX trip.

50 **ID: 1271568** **Points: 1.00**

Given:

- Unit 1 was at full power for the past 9 months, normal alignment, MOL.
- An RCS LOCA occurred.
- A reactor trip and SI were initiated and all equipment operated as designed.
- Plant conditions indicated a small break LOCA.
- The crew is currently performing 1BwEP ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

Adequate shutdown margin must be verified because of which of the following major evolutions in 1BwEP ES-1.2?

A. The RCS cooldown to 200°F.

B. The RCS depressurization.

C. The restarting of an RCP.

D. The shutdown of the RH pumps from ECCS injection.

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: #50
History: New for 2016 Bwd NRC exam
RO level
High Cog

K/A: EPE 009 Small Break LOCA
A2.32 Ability to determine or interpret the following as they apply to a small break LOCA: SDM
RO 3.2 SRO 3.6

The question meets the K/A, requires examinee knowledge of what evolutions are performed during a small break LOCA that affect SDM.

TIER:	1
GROUP:	1
Task No:	R-RK-005
Obj No:	3B.GP-02-1-2

10CFR55 Link: 10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
Technical Reference with Revision Number: 1BwEP ES-1.2 rev203

Answer Explanation: Cooling down the RCS will add positive reactivity to the core with a negative MTC, thereby reducing shutdown margin. 1BwEP ES-1.2 contains a note just prior to step 8, Initiating cooldown to 200°F that reads "Shutdown Margin should be monitored during the RCS cooldown.

Choice A is correct, see explanation above.
Choice B is incorrect, but plausible because RCS depressurization is a major evolution in 1BwEP ES-1.2 which causes a rise in ECCS cooling flow, however the depressurization will cause SDM to rise as the higher cooling flow is supplied with highly borated water from the RWST.
Choice C is incorrect, but plausible because restarting an RCP is a major evolution in 1BwEP ES-1.2, however it is not a concern for SDM, the flow of ECCS into the RCS will provide higly borated water in the leg being started. This is a concern during a SG tube rupture with backflow occurring due to the possibility of sending a large volume of unborated secondary water into the core.
Choice D is incorrect, but plausible because securing ECCS pumps reduces the rate of borated water being added to the core. However, the RCS pressure in a small break LOCA is above the RH pump shutoff head in ECCS injection mode. Stopping the RH pumps is not a concern for SDM because the pumps will not be injecting into the RCS when running.

51 **ID: 1269477** **Points: 1.00**

Given:

- Unit 1 has experienced a Large Break LOCA combined with a Loss of Offsite Power from 100% power.
- ALL systems operate as designed.

With the above conditions, which of the following describes the ORDER that the ECCS pumps will start and provide flow to the RCS?

A. 1. RH
2. CV
3. SI

B. 1. CV
2. RH
3. SI

C. 1. CV
2. SI
3. RH

D. 1. SI
2. CV
3. RH

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #51 **History:** Bank from Bwd 2011 NRC exam RO level High Cog **K/A:** EPE 011 Large Break LOCA K2.02 Knowledge of the interrelations between the following and a Large Break LOCA: pumps RO 2.6 SRO 2.7 The question meets the K/A, requires examinee knowledge of interrelation between a large break LOCA and ECCS pumps. TIER: 1 GROUP: 1 Task No: R-EF-001 Obj No: 3C.EF-01-B **10CFR55 Link:** 10CFR55.41(b)(8) Components, capacity, and functions of emergency systems. **Technical Reference with Revision Number:** I1-EC-XL-01 rev. 5c Bwd ILT ECCS lesson plan page 44 Bwd Big Note DG-2 rev. 8 **Answer Explanation:** With a loss of Offsite power and SI actuated, the ECCS pumps will restart on the ESF sequencer. The sequencing order and timing for the ECCS pumps are CV at 0 sec., SI at 5 sec. and RH at 10 sec. 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.

52 | **ID: 1268508** | **Points: 1.00**

Given the following sequence of events:

- Unit 1 was in Mode 1, normal alignment.
- A slave relay malfunction caused 1CC9414, CC FROM RC PUMPS ISOL VLV, to CLOSE and it will NOT re-open.

With the current CC valve alignment, the U-1 RCPs...

A. can remain running if RCP seal injection flows are maintained in their normal operating range.

B. MUST be tripped due to a loss of CC to ONLY the motor bearing oil cooler heat exchangers.

C. MUST be tripped due to a loss of CC to ONLY the thermal barrier heat exchangers.

D. MUST be tripped due to a loss of CC to the motor bearing oil cooler heat exchangers AND thermal barrier heat exchangers.

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #52
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: APE 015 Reactor Coolant Pump malfunctions
K3.02 Knowledge of the reasons for the following responses as they apply to the Reactor Coolant Pump Malfunctions: CCW lineup and flow paths to RCP oil coolers
RO 3.0 SRO 3.1

The question meets the K/A because the candidate must know why the RCP would need to be tripped with the given event.

TIER:	1
GROUP:	1
Task No:	R-OA-027
Obj No:	

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Technical Reference with Revision Number: Lesson Plan I1-RC-XL-02 rev 5b page 21, 1BwOA PRI-6 erv 108 page 8.

Answer Explanation: If an inadvertent Phase B isolation occurs, alarm response procedure BwAR 1-5-A7 directs the operator to reset the Phase B and re-open the CC valves. Subsequently the BwAR directs a reactor trip and RCP trips if CC cannot be established to the motor bearing heat exchangers. CC flow to the RCPs has a common supply line for both the motor bearing heat exchangers and the thermal barriers. The return lines are separate and 1CC9414 is the isolation for the return from the motor bearing heat exchangers.
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.

53 **ID: 1268509** **Points: 1.00**

In 1BwOA PRI-15, LOSS OF NORMAL CHARGING, the operator is directed to check the following indications:

- RCP No.1 Seal Leakoff Flow
- Charging Pump Flow
- Charging Pump Discharge Pressure
- Charging Pump Amps

The reason these parameters are monitored is to determine if there are indications of a...

A. seized pump shaft.

B. gas bound pump.

C. sheared pump shaft.

D. degraded pump impeller.

Answer: B

EXAMINATION ANSWER KEY

2016 NRC

Answer Explanation
Bwd 2016 NRC Exam Question: #53 **History:** New for Bwd 2016 NRC exam RO level Low Cog **K/A:** APE 022 Loss of Reactor Coolant Makeup K3.02 Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup: Actions contained in SOPs and EOPs for RCPs, loss of makeup, loss of charging, and abnormal charging RO 3.5 SRO 3.8 The question meets the K/A because the candidate must know why the CV pump/RCP parameters are checked in 1BwOA PRI-15. TIER: 1 GROUP: 1 Task No: R-OA-031 Obj No: T.OA23A-04 **10CFR55 Link:** 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility. **Technical Reference with Revision Number:** 1BwOA PRI-15 rev1, page 6 **Answer Explanation:** The listed parameters are checked for fluctuating indication in 1BwOA PRI-15. The reason they are checked is to determine if gas binding of the previously operating pump occurred. The procedure has a different mitigation strategy for a gas bound pump vs. any of the other distractors. For a gas bound pump, the procedure directs the standby pump to be vented prior to starting. Whereas, any of the distractors would only require the standby pump to be started (without prior venting). 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.

54 **ID: 1268510** **Points: 1.00**

Given:

- Unit 2 is in Mode 5.
- All systems are normally aligned.
- 2A RH pump is running in the shutdown cooling mode per BwOP RH-6.
- 2RH606, 2A RH HX OUTLET FCV, is throttled 10% open, with stable RCS temperature.
- 2RH618, 2A RH HX BYPASS FCV, is in AUTOMATIC.
- Pressurizer level is approx. 50% and being MANUALLY raised toward a solid condition.

The following occurs:

- A 100 gpm leak develops immediately DOWNSTREAM of the 2A RH Heat Exchanger.
- Pressurizer level is slowly lowering.

Based on the above conditions and assuming NO operator actions, 15 minutes later RCS temperature is now… (assume decay heat load remains constant)

A. HIGHER, and 2FI-618, PP 2A DISCH FLOW, is THE SAME.

B. HIGHER, and 2FI-618, PP 2A DISCH FLOW, is LOWER.

C. THE SAME, and 2FI-618, PP 2A DISCH FLOW, is LOWER.

D. THE SAME, and 2FI-618, PP 2A DISCH FLOW, is THE SAME.

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: #54
History: Bank from 2013 Bwd Cert exam
RO level
High Cog

K/A: APE 025 Loss of Residual Heat Removal System
A1.01 Ability to operate and/or monitor the following as they apply to the Loss of Residual Heat Removal System: RCS/RHRS cooldown rates
RO 3.6 SRO 3.7

Question meets KA - question requires examinee to predict impact of RCS leak on RH operation and RCS temperature.

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

TIER:	1
GROUP:	1
Task No:	R-RH-004
Obj No:	3D.OA-09-D

Technical Reference with Revision Number: Lesson plan I1-RH-XL-01 rev 5d, page 30, big note rh-1b rev0

Answer Explanation: A leak downstream of the RH HX would cause flow to initially drop but 2RH618 FCV would throttle open to maintain 3300 gpm return flow (indicated flow) to the RCS , Since less flow coming out of the RH HX would make it back to the RCS and more flow actually bypasses the RH HX, RCS temperature would rise.
A is correct, , see explanation above.
B is incorrect, see explanation above.
C is incorrect, see explanation above.
D is incorrect, see explanation above.

55 **ID: 1268529** **Points: 1.00**

Given:

- Unit 1 is in Mode 3 at NOP/NOT.
- Pressurizer pressure control is selected to channel 455/456 in automatic.
- PZR pressure channel 1P-457 is in TEST (all required bistables tripped).

Subsequently the following occurs:

- 1PT-458 fails LOW.
- SI ACTUATES and all equipment functions as designed.

5 minutes later, NO manual operator actions have been taken and the following conditions are noted:

- Tave is 555°F and slowly LOWERING.
- Pressurizer level is 35% and slowly RISING.
- Pressurizer pressure is 2280 psig and slowly RISING.
- 1PK-455A, MASTER PZR PRESS CONT, demand is 50%.

With the above conditions, PZR Backup Heaters will be ___(1)___ and PZR spray valves will be ___(2)___.

A. (1) Off
(2) Closed

B. (1) Off
(2) Open

C. (1) On
(2) Closed

D. (1) On
(2) Open

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #55 **History:** New for Bwd 2016 NRC exam RO level High Cog **K/A:** APE 027 Pressurizer Pressure Control System Malfunction K1.03 Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Latent heat of vaporization/condensation RO 2.6 SRO 2.9 The question meets the K/A by knowing the operational implications of a potential loss of saturation conditions in the pressurizer. Backup heaters are energized during pressurizer insurge conditions to add latent heat of vaporization and maintain saturation conditions. TIER: 1 GROUP: 1 Task No: R-OA-100 Obj No: T.OA11-25 **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** Big Note RY-2 rev10 and RY-3 rev11 PZR Level Control **Answer Explanation:** When SI actuates from Mode 3, NOP/NOT conditions, RCS temp will drop slowly due to the injection of ECCS flow and pressure will rise slowly due to the pressurizer bubble being compressed by an insurge. The pressurizer insurge of relatively cooler water (if allowed to continue long term) may have the effect of losing saturation conditions in the pressurizer. The PZR backup heaters are interlocked to turn on when PZR level reaches or exceeds 5% above program level. In this condition, program level will initially be at minimum (25%). Therefore, the backup heaters will be on with PZR level at 35%. Actual PZR pressure and master pressure controller demand will not demand heaters to be on and will demand PZR spray valves to be open. However, the 5% above program level interlock will override the heater demand and turn on the backup heaters. PZR sprays will not be open because the SI will cause a containment phase A isolation and isolate instrument air to containment failing the PZR spray valves closed. 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.

56 **ID: 1268531** **Points: 1.00**

RCS over pressurization is the PRIMARY concern during a/an ______(1)_______ event, and prompt operator mitigation actions may prevent RCS pressure from exceeding the TECH SPEC RCS PRESSURE SAFETY LIMIT of ___(2)___.

A. (1) Anticipated Transient Without Scram
(2) 3107 psig

B. (1) Anticipated Transient Without Scram
(2) 2735 psig

C. (1) Loss of Heat Sink
(2) 3107 psig

D. (1) Loss of Heat Sink
(2) 2735 psig

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #56
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: EPE 029 ATWS
Gen 02.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
RO 3.2 SRO 4.2

The question meets the K/A by requiring the examinee to know tech spec safety limits.

TIER:	1
GROUP:	1
Task No:	R-FR-018
Obj No:	S.TS1-03-D

10CFR55 Link: 10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
Technical Reference with Revision Number: Lesson plan I1-FR-XL-01 rev 8 page 2 and 3, Technical specification 2.0 Safety Limits
Answer Explanation: Because of the fission heat produced during an ATWS event, the primary concern is rising RCS temperature which then causes a rising RCS pressure that will exceed the design pressure of the RCS. In contrast, the primary concern during a loss of heat sink event is the fuel clad integrity. A loss of heat sink is much slower developing, and although RCS pressure will rise as temperature rises, the relief capabilities of the pressurizer will not be exceeded. The RCS pressure safety limit listed in Tech Specs is ≤2735 psig. The distractor of 3107 psig is the hydrostatic test pressure limit.

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.

57 **ID: 1268538** **Points: 1.00**

Given:

- A SGTR event occurred on Unit 1.
- Currently the crew is performing 1BwEP-3, STEAM GENERATOR TUBE RUPTURE, at step 6, Initiate RCS Cooldown.
- ALL INTACT SG NR levels are at 50% and stable with AF isolated prior to the RCS cooldown.
- The crew is about to re-initiate AF flow to the intact SGs and then commence an RCS cooldown at maximum rate with the steam dumps.

With the above conditions, regarding the INTACT SGs, how will the operator actions affect the INITIAL indicated SG NR level?

(Consider each action separately)

1. A rapid addition of AF flow will cause SG NR level to ____(1)____.
2. Rapidly opening the steam dumps will cause SG NR level to ____(2)____.

A. (1) Shrink
(2) Shrink

B. (1) Swell
(2) Swell

C. (1) Shrink
(2) Swell

D. (1) Swell
(2) Shrink

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #57 **History:** New for Bwd 2016 NRC exam RO level High Cog **K/A:** EPE 038 SGTR A02.05 Ability to determine or interpret the following as they apply to a SGTR: Causes and consequences of shrink and swell in S/Gs RO 2.8 SRO 2.9 The question meets the K/A by requiring the examinee to know the causes of shrink and swell in the SGs. TIER: 1 GROUP: 1 Task No: R-EP-012 Obj No: A.HT3-08-B **10CFR55 Link:** 10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics. **Technical Reference with Revision Number:** Lesson plan I1-HT-XL-03 rev3a page 17 and 18 **Answer Explanation:** Feed Water Induced Steam Generator Shrink occurs when a step change in feedwater flow and/or temperature occurs. Rapid addition of relatively cooler feedwater flow entering the SG downcomer region causes a sudden contraction or collapse of steam bubbles in the downcomer region. This causes a redistribution of the water mass in the downcomer region with less mass being located above the narrow range level transmitter's lower tap causing a "shrink" of indicated level. By contrast, rapidly opening the steam dumps would create steam demand and lower SG pressure. The lowering of pressure causes a sudden expansion or formation of steam bubbles in the tube area. This causes a redistribution of the water mass in the downcomer region with more mass being located above the narrow range level transmitter's lower tap causing a "swell" of indicated level. 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.

58 **ID: 1268545** **Points: 1.00**

Given:

- Unit 1 plant startup was in progress.
- A main steamline break then occurred.
- The crew has implemented 1BwFR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK CONDITION.
- During performance of 1BwFR-P.1, conditions requiring a RCS temperature soak were identified.

In accordance with 1BwFR-P.1, the temperature soak specifies that...

A. an immediate RCS cooldown CAN continue, BUT the cooldown rate is restricted to LESS than Tech Spec limits.

B. NO further RCS cooldown UNTIL an engineering evaluation is performed to determine restrictions.

C. NO further RCS cooldown for a specified time period, THEN the cooldown rate is restricted TO Tech Spec RCS cooldown limits.

D. NO further RCS cooldown for a specified time period, THEN the cooldown rate is restricted to LESS than Tech Spec RCS cooldown limits.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #58
History: Bank from Bwd EOPS
RO level
Low Cog

K/A: EPE 040 Steam Line Rupture
K01.01 Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture: Consequences of PTS
RO 4.1 SRO 4.4

The question meets the K/A by requiring the examinee to know the operational consequences of PTS conditions.

TIER:	1
GROUP:	1
Task No:	R-FR-022
Obj No:	T.FR04-02

10CFR55 Link: 10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
Technical Reference with Revision Number: 1BwFR-P.1 rev 201 step 24 page 21.
Answer Explanation: The overall mitigation strategy and step 24 of 1BwFR-P.1 list criteria for a soak and the restrictions there after. The restrictions are to not cool down for a one hour period and then restrict the cool down rate to 50°F/hr (vs. Tech Spec limit of 100°F/hr.). The restriction remains in effect for any subsequent procedure that may be applicable.

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.

59 **ID: 1268554** **Points: 1.00**

Given:

- Unit 1 is experiencing a loss of heat sink condition with the following plant conditions:

- 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, attempting 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 25%.

D. 1A CV pumps trips.

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #59 **History:** Bank from 2011 Bwd NRC Exam RO level High Cog **K/A:** APE054 Loss of Main Feedwater A01.04 Ability to operate and / or monitor the following as they apply to the Loss of Main Feedwater (MFW): HPI, under total feedwater loss conditions RO 4.4 SRO 4.5 The question meets the K/A, requires examinee to monitor HPI (CV pump operation) and apply it to loss of heat sink conditions (total loss of feedwater). TIER: 1 GROUP: 1 Task No: R-FR-029 Obj No: 4D.FR-08 **10CFR55 Link:** 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility. **Technical Reference with Revision Number:** 1BwFR-H1 rev 205 OAS page **Answer Explanation:** 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 is still below 2335#. This pressure is above normal operating pressure which makes it plausible. 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 criteria values of 27%. Choice D is incorrect, 1B CV pp would still be running.

60 **ID: 1268556** **Points: 1.00**

Given:

- Unit 1 has experienced an extended Loss of All AC Power.
- The crew is performing 1BwCA-0.0, LOSS OF ALL AC POWER, Attachment B, step 10, DEPRESSURIZE ALL INTACT SGs TO 260 PSIG.

With the above conditions, which of the following is the basis for depressurizing the SGs?

A. Maximize Aux. Feed flow rates.

B. Prevent lifting PZR safety valves.

C. Minimize SG tube differential pressure.

D. Minimize RCS inventory loss.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #60
History: Bank from 2011 Bwd NRC exam
RO level
Low Cog

K/A: EPE055 Loss of Offsite and Onsite Power (Station Blackout)
K03.02 Knowledge of the reasons for the following responses as they apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power.

The question meets the K/A, requires examinee knowledge of reasons for actions contained in the emergency procedure for loss of all AC power.

TIER:	1
GROUP:	1
Task No:	R-CA-009
Obj No:	3D.CA-01-C

10CFR55 Link: 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Technical Reference with Revision Number: WOG Background Document for CA-0.0 rev 203 page 106

Answer Explanation: Per the WOG background document for CA-0.0, the basis for depressurizing the SGs is to reduce RCS temperature and pressure to reduce RCP seal leakage and minimize RCS inventory loss. Per the ERG background document for ECA-0.0, operator action is taken to reduce SG pressures prior to allowing them to reach the SG safety setpoints to minimize the RCS inventory loss from imminent RCP seal failure.

Choice A is incorrect, the 1B AF pump is designed with adequate pump head pressure (990 gpm at 1450 psig) to overcome SG pressures up to the SG safety setpoints, therefore there is no reason to depressurize to raise flow.
Choice B is incorrect, with the RCP seal leakage anticipated during this event, pressurizer level and pressure will be dropping and not challenging the pressurizer safety setpoints.
Choice C is incorrect, during a SGTR event, steam is dumped to cooldown and eventually depressurize the RCS, however SG tube DP is not a primary concern during an loss of all AC event since the RCS pressure will be dropping due to the seal leakage.
Choice D is correct, see explanation above.

61	ID: 1268611	Points: 1.00

Given:

- Unit 1 was at 100% power, normal alignment.

Subsequently:

- A SAT fault results in a loss of offsite power.
- All equipment operates as designed.

Given the above conditions, with NO operator action over the next 30 minutes, SG level will...

A. rise slightly then stabilize back at normal level.

B. rise and then stabilize at a slightly higher than normal level.

C. slowly rise to the SG level P-14 setpoint.

D. slowly lower to the SG level LO-2 setpoint.

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: #61
History: Modified from Bwd 2014 NRC exam (RS1061-N14-27)
RO level
High Cog

K/A: 056 Loss of Off-site Power
A02.81 Ability to determine and interpret the following as they apply to the Loss of Offsite Power: S/G level meter scale and pressure gauge
RO 3.7 SRO 3.8

The question meets the K/A because the candidate must know the SG level response to a loss of offsite power event.

10CFR55 Link: 10CFR55.41(b)(7) Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

TIER:	1
GROUP:	1
Task No:	R-OA-005
Obj No:	T.OA04-12

Technical Reference with Revision Number: Lesson plan I1-FW-XL-01 rev 4 page 12 big note FW-2 rev 1 U1 SGWLC
Answer Explanation: On a loss of off-site power the ESF buses will de-energize momentarily until the DGs re-energize the ESF buses. The ESF loads will sequence on their respective buses including the 1A AF pump. The non-ESF buses that were previously supplied by the SATs will ABT to the UATs and the unit will not trip. When the 1A AF Pump starts, the SG levels will rise initially due to the added AF flow. Because the SG level control systems are level dominant, the SG water level will return to normal level with the Feed Reg Valves more closed because the 1A AF pump is supplying added AF flow, the Feed Reg Valves will throttle closed to compensate for the added AF flow.

A is correct. See explanation.
B is incorrect. Plausible if the examinee does not understand the SG water level system is level dominant. The additional input into the SG water level control system is Steam Flow Feed Flow mismatch. AF flow ties in after the FW flow is sensed, therefore if the SG was not level dominate, the SG water level would continue to rise because feedwater flow does not account for the additional AF flow.
C is incorrect. This is plausible if the examinee thinks that water level would continue to rise slightly with the added AF flow.
D is incorrect. This is plausible if the examinee believes the reactor will trip and feed water isolation will occur.

62 **ID: 1268612** **Points: 1.00**

Given:

- Unit 2 was at full power, normal alignment.

The following sequence of events then occurred:
- A loss of BOTH U-2 SX pumps occurred last shift.
- The crew SHUTDOWN and ISOLATED the 2A and 2C RCFC trains and cross tied the U-1 and U-2 SX systems.
- Per 2BwOA PRI-8, ESSENTIAL SERVICE WATER MALFUNCTION, the RO is monitoring containment temperatures by performing 2BwOSR 0.1-1,2,3, SHIFTLY DAILY OPERATING SURV.

The NSO observes the following RCFC temperature indications on 2PM06J:

- 2A INLET TEMP - 109°F
- 2A OUTLET TEMP - 82°F

- 2C INLET TEMP - 107°F
- 2C OUTLET TEMP - 78°F

- 2B INLET TEMP - 100°F
- 2B OUTLET TEMP - 75°F

- 2D INLET TEMP - 102°F
- 2D OUTLET TEMP - 75°F

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

A. 88.0 °F

B. 91.0 °F

C. 101.0°F

D. 104.5 °F

Answer: C

Answer Explanation

Bwd 2016 NRC Exam Question: #62
History: Modified from 2009 Bwd NRC exam (RS20103-N01)
RO level
High Cog

K/A: APE 062 Loss of Nuclear Service Water
Gen 02.12 Knowledge of surveillance procedures
RO 3.7 SRO 4.1
The question meets the K/A because the candidate must know how to properly calculate containment temperature for a surveillance procedure during a loss of SX event.

TIER:	1
GROUP:	1
Task No:	R-AM-006
Obj No:	S.VP1-06

10CFR55 Link: 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Technical Reference with Revision Number: 2BwOSR 0.1-1,2,3 rev 83 page 8 step F 5

Answer Explanation: 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 2A & C RCFCs are shut down, their inlet temps are not used in the average calculation.
Choice A is incorrect, this is average of 2B & 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 2A & C inlet temp should not be used.
Choice C is correct, see explanation above.
Choice D is incorrect, this is average of ALL inlet temps. 2A & C temps should not be used.

63 **ID: 1268629** **Points: 1.00**

Given the following sequence of events:

- Unit 1 was at 100% power.
- ESF bus 142 is de-energized due to a bus fault.

Subsequently:

- A Manual reactor trip and SI occurred 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 1PM05J/6J:

- RCS wide range pressure is 1700 psig and stable.
- 1A RH discharge flow is 0 gpm.
- 1A SI pump discharge flow is 200 gpm.
- High Head SI flow is 300 gpm.

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

A. 1RH8716A, RH HX 1A DSCH XTIE VLV.

B. 1SI8801A, CHG PMPS TO COLD LEGS INJ ISOL VLV.

C. 1SI8809A, RH TO COLD LEGS 1A & 1D ISOL VLV.

D. 1SI8835, SI PMPS TO COLD LEGS ISOL VLV.

Answer: D

Answer Explanation
Bwd 2016 NRC Exam Question: #63 **History:** Bank from 2009 Bwd NRC (a modified form of this question is on the 2016 Cert exam) RO level High Cog **K/A:** W/E04 LOCA Outside Containment K01.02 Normal, abnormal and emergency operating procedures associated with (LOCA Outside Containment) RO 3.5 SRO 4.2 The question meets the K/A because the candidate must determine success path in emergency procedure based upon MCR indications. TIER: 1 GROUP: 1 Task No: R-CA-004 Obj No: 4D.CA-03 **10CFR55 Link:** 10CFR55.41(b)(8) Components, capacity, and functions of emergency systems **Technical Reference with Revision Number:** Big note ECCS-1 rev 12 ECCS System **Answer Explanation:** 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 conclude 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. The bus 142 fault is in the stem to limit ECCS flows to one train for more accurate assessment capabilities of the individual pumps. Choice A is incorrect, RH pump flow is normal for current RCS pressure. Choice B is incorrect, CV pump flow is normal for current RCS pressure. Choice C is incorrect, RH pump flow is normal for current RCS pressure. Choice D is correct, SI pump flow is abnormal for current RCS pressure.

64	ID: 1268642	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, 1BwCA-1.1 requires that the NSO maintain a MINIMUM primary plant inventory of...

A. 4% in the Pressurizer.

B. 14% in the Pressurizer.

C. 31% in RVLIS HEAD region.

D. 15% in RVLIS PLENUM region.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #64
History: Modified from Bwd 2009 NRC Exam (RE1WE11-N01)
RO level
Low Cog

K/A: W/E11 Loss of Emergency Coolant Recirculation
A01.03 Ability to operate and / or monitor the following as they apply to the (Loss of Emergency Coolant Recirculation): Desired operating results during abnormal and emergency situations

The question meets the K/A, requires examinee knowledge of desired results associated with loss of emergency coolant recirc.

TIER:	1
GROUP:	1
Task No:	R-CA-003
Obj No:	3D.CA-02-B

10CFR55 Link: 10CFR55.41(b) (10) Administrative, normal, abnormal, and emergency operating procedures for the facility

Technical Reference with Revision Number: 1BwCA-1.1 rev 205 step 21 page 18

Answer Explanation: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, 4% pressurizer level is criteria for determining SI actuation criteria in 1BwEP ES-0.2.
Choice B is incorrect, 14% pressurizer level is criteria for determining if ECCS flow should be reduced in 1BwEP-0
Choice C is incorrect, 31% in the RVLIS head region would represent the first indication that reactor vessel inventory was not completely full.
Choice D is correct, see explanation above.

65	ID: 1268648	Points: 1.00

Given:

- Unit 1 at 100% power in a normal alignment.
- 1VI-MP006, MAIN GENERATOR 1 OUTPUT VARS, indicates 200 MVARS out.

Subsequently:

- A grid disturbance results in grid voltage DROPPING.
- TSO reports NO switchyard lines were lost and total grid reactive load did NOT change.

In response to this event, Unit 1 MVARS out …

A. lowered. To return MVAR load to a STABLE 200 MVARS OUT, the crew will take the VOLT adjuster control switch to RAISE.

B. did NOT change with the voltage regulator in Automatic. NO adjustment will be necessary.

C. rose. To return MVAR load to a STABLE 200 MVARS OUT, the crew will take the BASE adjuster control switch to LOWER.

D. rose. To return MVAR load to a STABLE 200 MVARS OUT, the crew will take the VOLT adjuster control switch to LOWER.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #65
History: Modified from Bwd 2013 NRC exam (RE10077-C03)
RO level
High Cog

K/A: APE 077 Generator Voltage and Electric Grid Disturbances
K02.07 Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following:Turbine / generator control
RO 3.6 SRO 3.7

Meets K/A, examinee must assess the effect of changing grid voltage on main generator reactive load and determine the appropriate voltage controller adjustment.

TIER: 1
GROUP: 1
Task No: R-MP-001
Obj No: S.MP2-05-A.
10CFR55 Link: 10CFR55.41(b)(5) Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
Technical Reference with Revision Number: Lesson plan I1-MP-XL-02MG rev 6 b page 36

Answer Explanation: A drop in grid voltage with no change in total reactive load causes a rise in reactive load output of the main generator, therefore MVARS out will rise. To restore reactive load to the original value, the operator must go to LOWER on the voltage regulator control switch. The base adjuster will have no effect with the voltage regulator in automatic. Control is shifted to the Base Adjuster if/when the automatic voltage regulator is taken to "off".

A is incorrect. See explanation.
B is incorrect. See explanation.
C is incorrect. See explanation.
D is correct. See explanation.

66 **ID: 1268688** **Points: 1.00**

Given:

- Unit 1 reactor power is 65% and stable.
- Main Generator load is 800 MW and stable.
- 1C zone Main Condenser pressure is 1.6 inches 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 air leak.
- The RO 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 that the unit will first reach the unacceptable region of Figure 1BwOA SEC-3-1?

(assume the Load DROP rate remains constant and the rate of Main Condenser pressure RISE remains constant throughout the event)

A. 1119.

B. 1124.

C. 1127.

D. 1132.

Answer: C

Answer Explanation

Provide reference: 1BwOA SEC-3 rev 106 page 12 of 14, Figure 1BwOA SEC-3-1

Bwd 2016 NRC Exam Question: #66
History: Bank from Bwd 2009 NRC exam
RO level
High Cog

K/A: Gen 2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.

The question meets the K/A, requires examinee ability to use a graph to determine conditions requiring unit trip as applied to loss of condenser vacuum.

TIER:	3
GROUP:	1
Task No:	R-AM-022
Obj No:	A.MX6-01-A

10CFR55 Link: 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Technical Reference with Revision Number: 1BwOA SEC-3 rev 106 page 12

Answer Explanation:
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).

67	ID: 1268708	Points: 1.00

Given:

- Unit 1 is performing an INITIAL power ascension following a refueling outage.
- Current reactor power level is 50% and slowly rising.

With the above conditions, Rod Control will be maintained in...

A. AUTOMATIC to ensure Tave/Tref deviations are maintained within 1BwGP 100-8, GENERIC REACTOR CONTROL GUIDANCE, limits.

B. AUTOMATIC to ensure Shutdown Margin is maintained in accordance with Tech Spec Rod Insertion Limits.

C. MANUAL to ensure Axial Flux Distribution limits in the COLR are NOT exceeded.

D. MANUAL to ensure fuel conditioning rod withdrawal rate limits are NOT exceeded.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #67
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: Gen 2.1.32 Ability to explain and apply system limits and precautions.

The question meets the K/A, requires examinee ability explain limitations of rod control.

TIER:	3
GROUP:	1
Task No:	R-GP-007
Obj No:	3B.GP-02-A-3

10CFR55 Link: 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Technical Reference with Revision Number: 1BwGP 100-3 rev 66, page 9

Answer Explanation:
Per 1BwGP 100-3, Limitation E.1.c.9, Do not use automatic rod control during the initial power ascension to avoid exceeding fuel conditioning rod withdrawal rate limits. During the initial power ascension, rod withdrawal is limited to 3 steps per hour until rods are withdrawn to at least 227 steps at no less than 98% power. The distractors are all limits than need to be adhered to, however, none are referred to in the BwGP procedures as requiring a specific rod control mode of operation.
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.

68 **ID: 1268709** **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, and 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
Bwd 2016 NRC Exam Question: #68 **History:** Bank from Bwd 2009 NRC exam RO level High Cog **K/A:** Gen 2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. RO 3.1 SRO 4.2 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. TIER: 3 CATEGORY: 2 Task No: R-AP-001 Obj No: 3C.AP-01-B **10CFR55 Link:** 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility **Technical Reference with Revision Number:** LCO 3.8.1 amendment 134, page 3.8.1-1, BIG NOTE AC-7 rev8 **Answer Explanation:** 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 based upon objectives in the ILT program). 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 LCO 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 incorrect, 1572 is SAT feed to non-ESF bus 157. No LCO is required.

69	ID: 1269483	Points: 1.00

Given:

- Unit 1 is preparing for a reactor startup following a refueling outage.
- An NSO has been assigned the task of performing low power physics testing.

Prior to performing the low power physics test, the minimum required briefing level for this task is a...

A. Standard Pre-Job Briefing.

B. Tailored Pre-Job Briefing.

C. Heightened Level of Awareness Briefing.

D. Infrequent Plant Activity Briefing.

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #69
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: Gen 2.2.07 Knowledge of the process for conducting special or infrequent tests.
RO 2.9 SRO 3.6

The question meets the K/A, requires examinee knowledge of process for conducting infrequent tests.

TIER:	3
CATEGORY:	2
Task No:	R-AM-133
Obj No:	3E.AM-133

10CFR55 Link: 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility

Technical Reference with Revision Number: HU-AA-1211 rev 11 page 11

Answer Explanation: HU-AA-1211 section 4.1.6 specifically designates low power physics testing as requiring an IPA briefing. All of the briefing level choices contain attributes listed in HU-AA-1211 that are common to low power physics testing. To correctly answer the question the candidate must either have the numerous examples of activities in HU-AA-1211 memorized, or have an operational understanding of what "low power physics testing" consists of.
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.

70 **ID: 1268711** **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 570.0 °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. step IN at 48 steps/minute.

B. step IN at 8 steps/minute.

C. NOT step.

D. step OUT at 8 steps/minute.

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #70
History: Modified from Bwd 2009 NRC Exam
RO level
High Cog

K/A: Gen 2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

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 effect of taking rods to auto.

TIER:	3
CATEGORY:	2
Task No:	R-RD-001
Obj No:	3C.RD-02-A

10CFR55 Link: 10CFR55.41(b) (6) Design, components, and function of reactivity control mechanisms and instrumentation.

Technical Reference with Revision Number: Big note RD-2 rev 6

Answer Explanation: Reactor 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.
Choice A is incorrect, direction is correct however, speed is setpoint for manual rod control.
Choice B is correct, see explanation above
Choice C is incorrect, would be correct if rod control used average Tave.
Choice D is incorrect, would be correct if rod control used auctioneered low Tave.

EXAMINATION ANSWER KEY

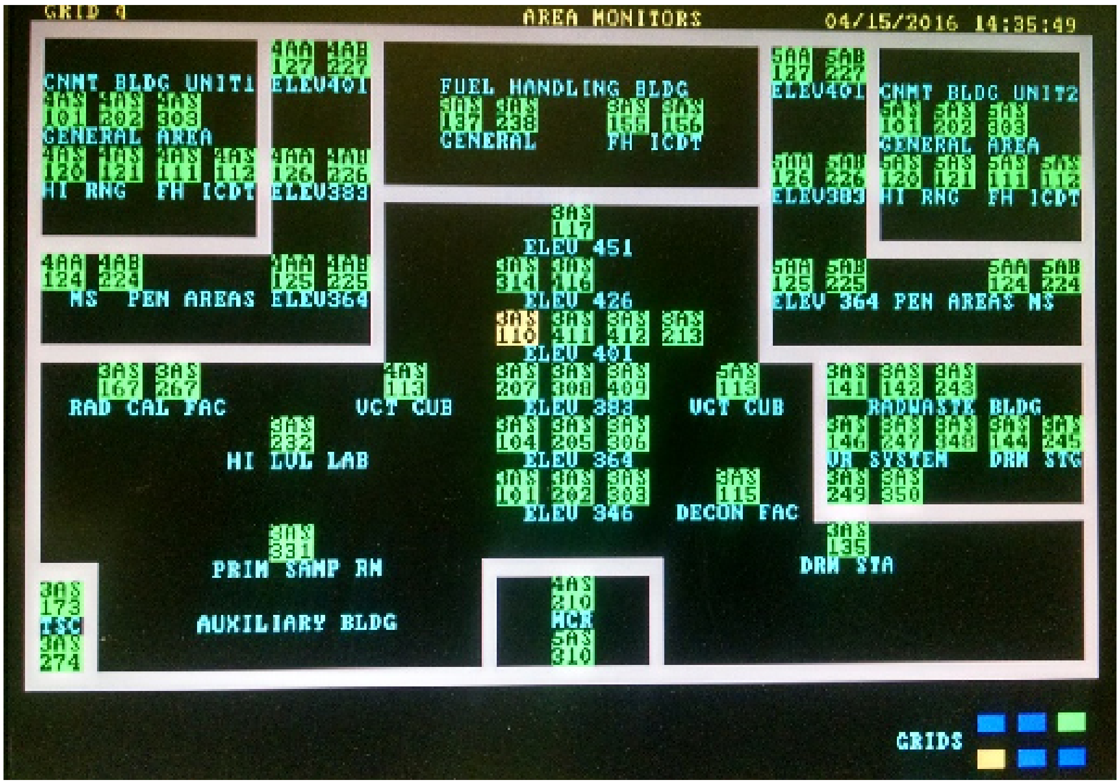
2016 NRC

71	ID: 1268728	Points: 1.00

Given the following plant conditions on Unit 1:

- An audible alarm on the RM-11 is received.
- The RO notes the below indications.

Which of the following plant evolutions (based on plant location), would be most likely to cause the alarm? (Analyze each evolution separately and assume each evolution causes local radiation levels to rise above the alert setpoint on the nearest area detector on the same elevation)



A. A reactor coolant filter is being removed from its filter vault.

B. The Hi Level Spent Resin Storage Tank is being de-watered.

C. A radwaste vendor demineralizer resin bed is being transferred to a bulk radwaste liner.

D. Radiography of the U-1 CC surge tank vent valve line.

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: #71
History: New for Bwd 2016 NRC exam
RO level
High Cog

K/A: Gen 2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities
RO 3.4 SRO 3.8

The question meets the K/A, requires examinee knowledge of locations associated with localized elevated rad conditions.

TIER:	3
CATEGORY:	3
Task No:	R-AR-002
Obj No:	S.AR1-09

10CFR55 Link: 10CFR55.41(b) (11) Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Technical Reference with Revision Number: BwAR 4-0AR060J rev 2

Answer Explanation: The area rad monitor channel 3AS110 in alert status is located in the aux bldg, 401' elevation on the Unit 1 side. This is the general area where RC filter vaults are opened and the filters removed for changing. The distractors would alarm different detector channels on the same RM-11 Grid 4 screen.

Choice A is correct, , see explanation above.
Choice B is incorrect, the Hi level spent resin storage tank is in a shielded room on the unit 2 side of 426' elevation.
Choice C is incorrect, the radwaste vendor demin and bulk resin liner loading area is on 401" elevation in the Solid Radwaste Bldg. This is the same elevation however, the radwaste building detector icons are grouped in a separate section on Grid 4 of the RM-11.
Choice D is incorrect, the CC surge tank vent line is on the 426' elevation (above the filter changing area). This can easily be confused with the alarming detector if the candidate associates the RM-11 icons with the elevation labeling above the icon (vs. the elevation labeling below the icon).

72 **ID: 1268736** **Points: 1.00**

Given:

- Both units are at full power, normal alignment.
- The 0A VC train is in normal operation per BwOP VC-1, STARTUP OF CONTROL ROOM HVAC SYSTEM.
- The 0B VC train is in standby.

The following then occurs:

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

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

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

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

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

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

Answer: A

EXAMINATION ANSWER KEY

2016 NRC

Answer Explanation
Bwd 2016 NRC Exam Question: #72 **History:** Bank from Bwd 2011 NRC exam RO level High Cog **K/A:** Gen 2.3.5 Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc. RO 2.9 SRO 2.9 The question meets the K/A, requires examinee ability to use radiation monitoring systems. TIER: 3 GROUP: 3 Task No: R-AR-002 Obj No: 4C.AR-02 **10CFR55 Link:** 10CFR55.41(b)(7) Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. **Technical Reference with Revision Number:** Bwd ILT LP I1-VC-XL-01 rev3a page 18 Bwd ILT LP I1-AR-XL-01 rev5b page 29 Big note VC-1 rev. 10 **Answer Explanation:** MCR rad monitor icons are green even when their sampled plenums are not online because the sample pumps will continuously sample plenums that have stagnant air flow. With the 0A VC system in normal alignment (outside air intake), the only rad monitors that would have MCR intake air flow through their respective intake plenum is the 0PR31J and 32J. Choice A is correct, see explanation above. Choice B is incorrect, 0PR33J and 34J sample the outside air intake from Unit 2 (0B train). Although the radiation event was in the Unit 2 turbine bldg, because 0B VC train was not running, this plenum would not experience intake air flow. Choice C is incorrect, 0PR35J and 36J sample the turbine bldg intake from Unit 1 (0A train). Although the radiation event was in the turbine bldg, this plenum would not experience intake air flow unless the 0A VC system was manually or automatically swapped to emergency mode. Choice D is incorrect, 0PR37J and 38J sample the turbine bldg intake from Unit 2 (0B train). Although the radiation event was in the Unit 2 turbine bldg, this plenum would not experience intake air flow unless the 0B VC system was manually started in emergency mode.

73 **ID: 1269488** **Points: 1.00**

Given:

- The unit 1 RO is responding to a reactor trip.
- 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, step 1 has just been completed by the RO.

With the above conditions, which of the following tasks would take precedent over continuing the Immediate Actions of 1BwEP-0?

A. The RO manually tripping RCPs after RCP trip criteria has been met.

B. The RO dispatching an EO to LOCALLY trip the PMG output breaker after a loss of DC bus 113 occurred.

C. The RO manually actuating Main Steam Isolation after a Main Steamline Break has been diagnosed.

D. The RO manually starting an ECCS pump that failed to automatically start after an RCS LOCA occurred.

Answer: C

Answer Explanation

Bwd 2016 NRC Exam Question: #73
History: New for Bwd 2016 NRC exam
RO level
Low Cog

K/A: 2.4.13 Knowledge of crew roles and responsibilities during EOP usage.
RO 4.0 SRO 4.6

The question meets the K/A, requires examinee ability to know their responsibility during EOP usage situation.

TIER:	3
CATEGORY:	4
Task No:	R-AM-133
Obj No:	3E.AM-133

10CFR55 Link: 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility.
Technical Reference with Revision Number: OP-BR-103-102-1002, section 4.8.2 rev. 5

Answer Explanation: OP-BR-103-102-1002, section 4.8.2 allows the RO to interrupt the immediate actions for a reactor trip after step 1 is complete to actuate MSI if a main steamline break has occurred. This preemptive measure is allowed for the safety of plant personnel. The distractors are all actions that the RO will take promptly for each given event, however none are intended to interrupt the immediate actions of 1BwEP-0.

Choice A is incorrect, tripping RCPs when trip criteria is met is on the OAS of 1BwEP-0, but is not an exception to the immediate actions.
Choice B is incorrect, dispatching an EO to trip the PMG output breaker is step 2 of 1BwOA ELEC-1, immediately after trip the reactor and performing immediate actions of 1BwEP-0. It is not an exception to the immediate actions.
Choice C is correct, see explanation above.
Choice D is incorrect, starting ECCS equipment that failed to start is an action that can be taken as a prudent operator action per BwAP 340-1 or in step 6 RNO of 1BwEP-0. It is not an exception to the immediate actions.

74 **ID: 1268770** **Points: 1.00**

Given:

- Unit 1 was at full power, normal alignment.
- A condition occurs requiring a reactor trip when an automatic reactor trip fails to happen.
- The RO was NOT able to trip the reactor from either of the main control room reactor trip switches.
- The Unit 1 main turbine was tripped.
- Control rods are in automatic and stepping in at 24 steps per minute.
- Current reactor power is 75% and slowly dropping.

With the above conditions, and in accordance with 1BwFR-S-1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, the RO will...

A. allow rods to continue to insert in automatic.

B. place the ROD BANK SELECT switch in MAN and insert rods.

C. place the ROD BANK SELECT switch in CBD and insert rods.

D. place the ROD BANK SELECT switch in SBD and insert rods.

Answer: B

Answer Explanation
Bwd 2016 NRC Exam Question: #74 **History:** New for Bwd 2016 NRC exam RO level Low Cog **K/A:** 2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. RO 4.6 SRO 4.0 The question meets the K/A, requires examinee ability to know the immediate actions required to control the control rod system during an ATWS. TIER: 3 CATEGORY: 4 Task No: R-FR-018 Obj No: T.FR01-02 **10CFR55 Link:** 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility. **Technical Reference with Revision Number:** 1BwFR-S-1 rev 201, page 2 **Answer Explanation:** Per immediate action step 1 RNO of 1BwFR-S-1, control rods will be allowed to step in in automatic until speed drops below 48 steps per minute. 48 steps per minute is the speed that the Rod control system will insert Rods in the manual position. The Rod control system should be taken to manual and the RO should insert Rods in manual to continue the fastest rate of negative reactivity insertion. By inserting the control banks in the manual position, they are inserted with overlap allowing all control banks to be inserted without having to change the bank selector switch position again. The overlap also allows two banks to step in at the same time providing the intended fastest insertion of negative reactivity. Choice A is incorrect, see explanation above. Allowing Rod control to insert Rods automatically is plausible, this is the preferred method until automatic rod speed drops below 48 steps per minute. Choice B is correct, see explanation above. Choice C is incorrect, see explanation above. This is plausible because CBD is the first CB that would insert followed by CBC, however in this position only control bank D inserts and then the next bank would have to be selected. Choice D is incorrect, see explanation above. This is plausible because placing rods in SBD is a required action for inadvertent rod motion. This position will insert a shutdown bank of control rods but a different position will have to be selected to continue adding negative reactivity once this bank is inserted.

75	ID: 1268771	Points: 1.00

The Braidwood Emergency Procedures are organized into two complimentary categories. These include EVENT based responses of ____(1)____ and Fission Product Barrier based responses of ____(2)____.

A. (1) BwEPs
(2) BwCAs and BwFRs

B. (1) BwFRs
(2) BwEPs and BwCAs

C. (1) BwEPs and BwCAs
(2) BwFRs

D. (1) BwEPs and BwFRs
(2) BwCAs

Answer: C

Answer Explanation
Bwd 2016 NRC Exam Question: #75 **History:** New for Bwd 2016 NRC exam RO level Low Cog **K/A:** 2.4.05 Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions RO 3.7 SRO 4.3 The question meets the K/A, requires examinee knowledge of how emergency procedures were developed and organized by the Westinghouse Owners Group. TIER: 3 CATEGORY: 4 Task No: R-AM-022 Obj No: T.EP00-07 **10CFR55 Link:** 10CFR55.41(b)(10) Administrative, normal, abnormal, and emergency operating procedures for the facility. **Technical Reference with Revision Number:** Lesson plan I1-EP-XL-00 rev 8 page 9. Westinghouse Owners Group Emergency Response Guidelines Executive Summary. **Answer Explanation:** The concept of Westinghouse Owners Group Emergency Response Guidelines are to utilize two complementary and interrelated guideline subsets (one event related and one function (fission product barrier) related). This is explained in detail in the Westinghouse Owners Group Emergency Response Guidelines Executive Summary. 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.

76 **ID: 1267649** **Points: 1.00**

Operating within Tech Spec limiting conditions for operations (LCOs), 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

Bwd 2016 NRC Exam Question: #76
History: Bank from Bwd 2009 NRC exam
SRO level
Low Cog

K/A: 035 Steam Generator
Gen 02.22 Knowledge of limiting conditions for operations and safety limits.

RO 4.0 SRO 4.7
The question meets the K/A, requires examinee knowledge of safety limits found in the bases for LCO 3.4.16 RCS specific activity.

TIER:	2
GROUP:	2
Task No:	S-TS-006
Obj No:	8E.TS-006

10CFR55 Link: 10CFR55.43(b) (2)

Technical Reference with Revision Number:
LCO 3.4.16 bases rev 81 page B 3.4.16-1
10 CFR 50.67
ILT lesson plan I1-BZ-XL-01 rev. 2 page 28 and 30

SRO Justification:
The question is SRO level because requires knowledge of the Tech Spec bases of LCO 3.4.16.
Answer Explanation:
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.

77	ID: 1267650	Points: 1.00

Given:

- Unit-1 was shutdown 5 DAYS AGO and is currently in Mode 6.
- Fuel moves are in progress inside containment.
- A fuel handing malfunction results in slightly elevated radiation levels in containment.

With the above conditions, the 1AR011/012J, Containment Fuel Handling Incident Radiation Monitors, actuation function __(1)__ required to be OPERABLE per Tech Specs because ___________(2)___________.

A. (1) IS
(2) the potential for radioactive releases in this condition may EXCEED post-accident offsite doses being maintained within the limits of 10CFR50.67.

B. (1) IS
(2) each penetration providing access from the containment atmosphere to the outside atmosphere MAY be open and operator action CANNOT ensure post-accident offsite doses are maintained within the limits of 10CFR50.67.

C. (1) is NOT
(2) each penetration providing access from the containment atmosphere to the outside atmosphere MUST be closed by a manual or automatic isolation valve or blind flange.

D. (1) is NOT
(2) the potential for radioactive releases is minimized and operator action is sufficient to ensure post-accident offsite doses are maintained within the limits of 10CFR50.67.

Answer: D

Answer Explanation
Bwd 2016 NRC Exam Question #77 **History:** New for Bwd 2016 NRC exam SRO level Low Cog **K/A:** 034 Fuel handling K3.01 Knowledge of the effect that a loss or malfunction of the Fuel Handling System will have on the following: Containment ventilation RO 2.4 SRO 2.9 The question meets the K/A, requires examinee knowledge of how containment ventilation system would be affected during an incident involving the fuel handling equipment system that led to elevated radiation in containment. TIER: 2 GROUP: 2 Task No: S-TS-006 Obj No: S.FH1-10-A **10CFR55 Link:** 10CFR55.43(b)(2)Facility operating limitations in the technical specifications and their bases **Technical Reference with Revision Number:** B 3.3.6 REV 97 page B 3.3.6-4 **SRO Justification:** The question is SRO level because requires knowledge of the Tech Spec bases of LCO 3.3.6. **Answer Explanation:** Per LCO 3.3.6 bases, While in MODES 5 and 6 without fuel handling in progress or when moving fuel in containment that is not RECENTLY IRRADIATED FUEL, the containment ventilation isolation instrumentation need not be OPERABLE since the potential for radioactive releases is minimized and operator action is sufficient to ensure post-accident offsite doses are maintained within the limits of 10CFR50.67. 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.

78 **ID: 1268851** **Points: 1.00**

Given:

- Unit 1 is at full power, normal alignment.
- 1BwOS MS-Q1, MAIN STEAM DUMP VALVE STROKE SURVEILLANCE, is in progress.
- During the stroke test of 1MS004C Steam Dump Valve , the valve stuck OPEN and would not re-close.
- The US directed the crew to MAINTAIN 1MS003C, Steam Dump Valve Upsteam Manual Isolation Valve CLOSED and complete the surveillance for all the other steam dump valves.
- 1BwOS MS-Q1 has been completed (with 1MS003C left CLOSED) and it was noted the acceptance criteria was not met for 1MS004C.

Currently, with 1MS003C closed, the 1MS004C will be unavailable to respond as expected to a steam dump demand signal of ___(1)___ and the US will track the position of 1MS003C in the __________(2)__________.

A. (1) 0% to 50%
(2) Equipment Status Tag (EST) Log

B. (1) 0% to 50%
(2) Degraded Equipment Log (Dequip)

C. (1) 51% to 100%
(2) Equipment Status Tag (EST) Log

D. (1) 51% to 100%
(2) Degraded Equipment Log (Dequip)

Answer: C

Answer Explanation

Bwd 2016 NRC Exam Question: #78
History: New for Bwd 2016 NRC exam
SRO level
High Cog

K/A: 041 Steam Dump System
A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Steam Valve Stuck Open

RO 2.8 SRO 3.1

The question meets the K/A because the candidate must predict the results of a loss of a steam dump and use procedures to control the effects of the malfunction.

TIER:	2
GROUP:	2
Task No:	S-OA-103
Obj No:	S.DU1-11

10CFR55 Link: 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Technical Reference with Revision Number: OP-AA-108-101, rev. 12, Big Note MS-4 rev.15

SRO Justification:
The question is SRO level because the admin processes for equipment control are SRO functions at Braidwood.

Answer Explanation: The steam dump stroke surveillance consists of locally isolating the steam dumps and then manually stroking them from the MCR. When a valve is stuck open the local manual isolation valve is maintained closed to prevent steam flow through the failed open valve. The steam dumps valves are divided into four groups that open sequencially (by group) as demand rises from 0% to 100%. The valves are grouped as follows:

	Group 1	Group 2	Group 3	Group 4
Valves	A, E & J	B, F & K	C,G & L	D, H & M
Demand signal	0%-25%	26%-50%	51%-75%	76%-100%

The C steam dump valve is in group 3 and opens on demand of 51%-75%. 0% to 50% is a plausible distractor considering the unusual grouping of the valves and that their MCR indication is laid out alphabetically (vs. by their respective group)
The correct tracking mechanism is the Equipment Status Log, which tracks equipment status tags that identify temporary status of equipment position to ensure configuration control per OP-AA-108-101. The degraded equipment log is a plausible distractor as it would be used for a similar purpose if the malfunction was on an SSC that is safety related.

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.

EXAMINATION ANSWER KEY

2016 NRC

79	ID: 1267661	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 CC flow to the RCPs and high bearing 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 available.
- CETCs are 1235°F and slowly rising.
- Containment is ADVERSE.
- SG NR Levels are as follows:

1A	18%
1B	21%
1C	24%
1D	28%

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

A. opening both Pressurizer PORVs.

B. remain at step 16 until support conditions are established, then start the 1D RCP.

C. immediately start the 1D RCP.

D. transition to SACRG-1, SEVERE ACCIDENT CONTROL ROOM GUIDELINE INITIAL RESPONSE.

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: #79
History: Bank from 2009 Bwd NRC exam
SRO level
High Cog

K/A: 003 RCPs
A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems

RO 2.7 SRO 3.1

The question meets the K/A, requires examinee to use the procedure to mitigate the consequences of RCPs with bearing temperature problems.

TIER:	2
GROUP:	1
Task No:	S-FR-009
Obj No:	7D.FR-002-A

10CFR55 Link: 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Technical Reference with Revision Number: 1BwFR-C.1 rev 200 page 27

SRO Justification:
The question is SRO level because it requires assessment of conditions and direct appropriate MCR actions and level of detail beyond just the note and overall mitigating strategy.
The question is SRO level because in the accident conditions described, the SRO would be directing the crew operations with a procedure step critical to reactor safety, especially when the action of starting an RCP would in all likeliness "sacrifice" the RCP without an adequate heat sink to mitigate the inadequate core cooling.

Answer Explanation: 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. In addition, if no SG NR level is above 31% (adverse containment) then no RCP is started. This is a concern for creep rupture failure mechanism of SG tubes. This is a decision point for implementing a section of the procedure. The RNO of step 16.b. implements the next mitigating strategy by restoring air to containment and opening both PZR PORVs in an attempt to reduce RCS pressure and allow for low head injection. SACRG-1 will not be transitioned to until the PZR PORVs have been opened.
Choice A is correct; RCP is NOT started due to inadequate SG level, even though per the note, support conditions are not necessary.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

80	ID: 1267662	Points: 1.00

Given:

- A SGTR is in progress on Unit 1.
- The crew is performing 1BwEP-3, STEAM GENERATOR TUBE RUPTURE, step 16, Depressurize RCS to Minimize Break Flow and Refill PZR.
- The crew opens BOTH pressurizer spray valves BUT one spray valve will only open to an INTERMEDIATE position (will NOT fully open).

Current plant conditions are:

- Containment Pressure is 0.3 psig and STABLE.
- Ruptured SG pressure is 1115 psig and STABLE.
- RCS pressure is 2000 psig and DROPPING at 50 psig/min.
- Pressurizer level is 10% and RISING at 10%/min.

(assume parameter trends remain constant)

With the above conditions, the PZR spray flow is ___(1)___ and the SRO will direct the crew to __________(2)___________.

A. (1) ADEQUATE
(2) CONTINUE the depressurization with current spray flow

B. (1) NOT adequate
(2) KEEP the spray valves open AND open ONE PZR PORV ONLY

C. (1) NOT adequate
(2) KEEP the spray valves open AND open BOTH PZR PORVs

D. (1) NOT adequate
(2) CLOSE the spray valves open AND open BOTH PZR PORVs

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #80
History: New for Bwd 2016 NRC exam
SRO level
High Cog

K/A: 010 Pressurizer Pressure Control System
A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP Spray valve failures
RO 3.9 SRO 3.9

The question meets the K/A, requires examinee to use the procedure to mitigate the consequences of inadequate spray flow for pressure control.

TIER:	2
GROUP:	1
Task No:	S-EP-058
Obj No:	T.EP04-08

10CFR55 Link: 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Technical Reference with Revision Number: 1BwEP-3 rev 208 page 18 and 23

SRO Justification:
The question is SRO level because it requires assessment of conditions and direct appropriate MCR actions and level of detail beyond just the note and overall mitigating strategy.

Answer Explanation: 1BwEP-3, step 16 directs the crew to spray with maximum available spray until depressurization criteria are satisfied. The criteria is:
- RCS pressure - LESS THAN RUPTURED SG(s) PRESSURE AND
- PZR level - GREATER THAN 14% (28% ADVERSE CNMT)OR
- PZR level - GREATER THAN 68% (62% ADVERSE CNMT)

With the given parameter trends, PZR level will reach the 68% stop criteria in 5.8 min. RCS pressure will reach ruptured SG pressure in 8.85 min. Therefore, the depressurization rate is NOT adequate because the PZR will fill to 68% before the RCS is depressurized far enough to stop the leak.
A note prior to step 16 directs the crew to step 17 if normal spray is NOT adequate. Step 17 directs opening only one PORV while continuing to spray.
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.

81 **ID: 1267717** **Points: 1.00**

Given:

- Unit 1 was at 100% power, normal alignment.
- The 1A SG FAULTED INSIDE containment.
- The crew tripped the reactor, actuated SI and entered the appropriate emergency procedures.
- Currently the crew is performing 1BwEP-2, FAULTED SG ISOLATION, at step 1, Check Main Steamline Isolation.
- ALL MSIVs are CLOSED.
- An audible alarm is then noted on the RM-11. (see below indication)

- Channel 4AA122 is reading 3.35 E-01 mr/hr.
- Channel 4AD422 is reading 2.49 E+01 mr/hr.

With the above conditions, ________(1)________ and the procedural flowpath going forward will be ___________(2)____________.

A. (1) 1A SG has also ruptured
(2) complete 1A SG isolation in 1BwEP-2, THEN transition to 1BwEP-3, SG TUBE RUPTURE.

B. (1) a DIFFERENT SG (NOT 1A) has ruptured
(2) complete 1A SG isolation in 1BwEP-2, THEN transition to 1BwEP-3, SG TUBE RUPTURE.

C. (1) 1A SG has also ruptured
(2) IMMEDIATELY transition to 1BwEP-3, SG TUBE RUPTURE.

D. (1) a DIFFERENT SG (NOT 1A) has ruptured
(2) IMMEDIATELY transition to 1BwEP-3, SG TUBE RUPTURE.

Answer: B

Answer Explanation
Bwd 2016 NRC Exam Question: #81 **History:** New for Bwd 2016 NRC exam SRO level High Cog **K/A:** 039 Main and Reheat Steam System A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Indications and alarms for main steam and area radiation monitors (during SGTR) RO 3.4 SRO 3.7 The question meets the K/A, requires examinee to use the procedure to mitigate the consequences of alarms for SGTR and select the procedural flowpath for mitigation. TIER: 2 GROUP: 1 Task No: S-EP-053 Obj No: T.EP03-03 **10CFR55 Link:** 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. **Technical Reference with Revision Number:** 1BwEP-2 rev 202 page 10 **SRO Justification:** The question is SRO level because it requires assessment of conditions and selection of procedural flowpath. **Answer Explanation:** With the current RM-11 indications, the 1D SG has a high alarm (red) and is ruptured. The 1A SG has an alert alarm, however this would be expected with a 1D SGTR because the MS line area rad monitors are in close proximity to each other (same MSIV room). 1BwEP-2 is a higher priority than 1BwEP-3 as the diagnostic steps are sequenced in 1BwEP-0. In 1BwEP-2 secondary radiation trends are checked AFTER the faulted SG is isolated. Therefore the correct procedure flowpath is to complete the isolation of 1A SG in 1BwEP-2 prior to transitioning to 1BwEP-3. 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.

82	ID: 1267745	Points: 1.00

Given the following plant conditions on BOTH Units:

- 1PR030J AND 2PR030J, Unit 1 AND Unit 2 Wide Range Gas Monitors, are INOPERABLE.
- 0VA019 and 0VA020, Unit 1 AND Unit 2 Vent Stack Effluent Flow Monitors, are UNAVAILABLE.
- 0VA01JA AND 0VA01JC Flow Indicators are INOPERABLE.

- 0VA01CA, Aux Building Supply Fan 0A, is RUNNING.
- 0VA02CA, Aux Building Exhaust Fan 0A, is RUNNING.
- ALL other Aux Building Supply and Exhaust Fans are STOPPED.
- 0VL02CA, Lab Exhaust Fan 0A, is RUNNING.
- 0VW03CA, Service Building and Solid Radwaste Fan 0A, is RUNNING.
- 0VF01CA, Aux Building Filtered Vent Fan 0A, is RUNNING.

- 1PB128, Unit 1 Vent Stack Effluent Low Range Gas, is 5.0 E-07 microCi/cc.
- 1PD428, Unit 1 Vent Stack Effluent High Range Gas is 4.5 E-03 microCi/cc..
- 2PB128, Unit 2 Vent Stack Effluent Low Range Gas, is 4.15 E-07 microCi/cc
- 2PD428, Unit 2 Vent Stack Effluent High Range Gas, is 9.5 E-04 microCi/cc.

Based on the above conditions, the station TOTAL release rate is...

A. 2.92 E+05 microCi/sec

B. 4.59 E+05 microCi/sec.

C. 5.20 E+01microCi/sec

D. 9.73 E+02 microCi/sec

Answer: B

Answer Explanation

Attach reference EP-AA-121-F-01, section 2 (pages 4 & 5), Manual Station Release Rate Determination

Bwd 2016 NRC Exam Question: #82
History: Bank from Bwd ILT NOPS
SRO level
High Cog

K/A: 073 Process Rad Monitoring
Gen04.21.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.
RO 4.0 SRO 4.6

The question meets the K/A because the candidate must know the logic of fan configurations and flows. Also the candidate must be able to use given PRM (vent stack rad monitor) readings and assess radioactivity release.

TIER:	2
GROUP:	2
Task No:	S.ZP-015
Obj No:	T.ZP1-24

10CFR55 Link: 10CFR55.41(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Technical Reference with Revision Number: Lesson plan I1-ZP-XL-01 rev2e, page 25
EP-AA-121-F-01, rev 004, section 2, page 4 and 5

SRO Justification:
This calculation is a SRO function because the SM is acting as the Shift Emergency Director during an accident.

Answer Explanation: From EP-AA-121-F-01, station release rate is the sum of both units highest _PR28J activity X vent stack flow X 472. With only 0VA02CA running, estimated unit 1 vent stack flow is flow is 214,230 (205,000 cfm + 9230) and unit 2 vent stack flow is 9,630 micro curie/sec (8,630 + 1,000). unit 1 release rate = 455,025 (214,230 X 4.5E-3 X 472) and unit 2 release rate = 4318 micro curie/sec (9630 X 9.5 E-4 X 472). Total of both units is 4.59 E+5.
Choice A is incorrect,
Choice B is correct, see explanation above.
Choice C is incorrect
Choice D is incorrect

83	ID: 1269033	Points: 1.00

The Tech Spec limits for the amount of stored diesel fuel oil that is required to be maintained on site is based upon having sufficient supply for each diesel generator to supply...

Note:

LOCA - Loss of Coolant Accident
LOOP - Loss of Off Site Power

A. 3 days of post design basis LOCA load demand.

B. 7 days of post design basis LOCA load demand.

C. 14 days of post LOOP shutdown load demand.

D. 30 days of post LOOP shutdown load demand.

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #83
History: Bank from Bwd 2011 NRC exam
SRO level
Low Cog

K/A: 064 Emergency Diesel Generator
Gen 2.1.32 Ability to explain and apply system limits and precautions.
RO 3.8 SRO 4.0
The question meets the K/A, requires examinee ability to explain system limits.

TIER:	2
Group:	1
Task No:	S-AM-003
Obj No:	7E.AM-003-A

10CFR55 Link: 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

Technical Reference with Revision Number: Tech Spec 3.8.3 bases rev. 26

SRO Justification: The question is SRO level because it requires knowledge of Tech Spec bases.

Answer Explanation: Tech Spec 3.8.3 bases states the DG are supplied with enough stored oil for 7 days of post design basis LOCA loads.
Choice A is incorrect, 3 days (72 hours) is action completion time for TS 3.8.1 qualified circuit.
Choice B is correct, see explanation above.
Choice C is correct, 14 days is action completion time for TS 3.8.1 DG.
Choice D is incorrect, 30 days is action completion time for TS 3.8.3 fuel oil properties out of tolerance.

84 **ID: 1267788** **Points: 1.00**

Given:

- Unit 1 is in Mode 2, reactor start-up in progress.
- As the RO is inserting control rods to level power, ONE CBD rod sticks and will NOT insert.
- The misaligned rod is 13 steps above its group demand.
- The crew enters 1BwOA ROD-3, DROPPED OR MISALIGNED ROD.

With the above conditions, the SRO will direct the crew to ________(1)_______ and then ________(2)_______.

A. (1) INSERT Control Banks (only) in reverse order
(2) TRANSITION to 1BwGP 100-2, REACTOR STARTUP, step 1

B. (1) INSERT ALL Control and Shutdown Banks in reverse order
(2) TRANSITION to 1BwGP 100-2, REACTOR STARTUP, step 1

C. (1) TRIP the reactor
(2) perform 1BwEP-0, REACTOR TRIP OR SI, CONCURRENTLY with 1BwOA ROD-3

D. (1) TRIP the reactor
(2) TRANSITION to 1BwEP-0, REACTOR TRIP OR SI, ONLY

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #84
History: New for Bwd 2016 NRC exam
SRO level
Low Cog

K/A: 000005 Inoperable/Stuck Control Rod
Gen 04.08 Knowledge of how abnormal operating procedures are used in conjunction with EOPs
RO 3.8 SRO 4.5

The question meets the K/A, requires examinee to know how the AOP is used (or not) in conjunction with EOPs.

TIER:	1
GROUP:	2
Task No:	S-OA-092
Obj No:	T.OA34-03

10CFR55 Link: 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations

Technical Reference with Revision Number: 1BwOA ROD-3 rev 106 page 4 RNO 3a

SRO Justification:
The question is SRO level because it requires detailed knowledge of 1BwOA ROD-3 beyond the overall mitigative strategy. While normally, reactor trip criteria in an AOP would be considered RO knowledge, the question stem puts the crew in an infrequent and unusual situation (in RNO column of procedure) which is beyond the expected scope of RO knowledge.

Answer Explanation: Per 1BwOA ROD-3, if the reactor is in Mode 2 and any rod misalignment is encountered, the reactor is tripped and 1BwEP-0 is transitioned to. Unlike many other AOPs, 1BwOA ROD-3 is not performed concurrently, because once the reactor is tripped, the mitigative actions in 1BwOA ROD-3 are no longer applicable.

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.

85 **ID: 1267828** **Points: 1.00**

Given:

- Unit 2 is at full power, normal alignment.
- A rising trend in 2PR027J, SJAE Radiation Monitor, is detected.
- The crew enters 2BwOA SEC-8, STEAM GENERATOR TUBE LEAK.
- The SRO is attempting to determine the leak size and the affected SG to evaluate for Tech Spec entry conditions.
- At step 4, Identify Leaking SG, the 2AR022J/023J, Main Steamline Radiation Monitors indicate NO discernable rise in radiation levels on any channel.

With the above conditions, an alternate method to evaluate WHICH SG is leaking is to ________(1)_______.
If Tech Spec entry conditions are met, the bases for MAXIMUM allowable Primary to Secondary leak rate is ________(2)_______.

A. (1) obtain activity levels of N-16 monitor(s) from chemistry.
(2) the leakage amount that can be detected within 1 hour.

B. (1) obtain activity levels of N-16 monitor(s) from chemistry.
(2) the leakage amount that would likely escalate to a tube rupture.

C. (1) check indications of 2PR08J, SG Blowdown Sample Radiation Monitor.
(2) the leakage amount that would likely escalate to a tube rupture.

D. (1) check indications of 2PR08J, SG Blowdown Sample Radiation Monitor.
(2) the leakage amount that can be detected within 1 hour.

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #85
History: New for Bwd 2016 NRC exam
SRO level
Low Cog

K/A: 000037 SG Tube Leak
Gen 04.46 Ability to verify that the alarms are consistent with the plant conditions.
RO 4.2 SRO 4.2

The question meets the K/A, requires examinee to know how to verify a rising rad monitor trend that will eventually lead to an alert alarm status.

TIER:	1
GROUP:	2
Task No:	S-OA-110
Obj No:	T.OA43-03

10CFR55 Link: 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

Technical Reference with Revision Number: 2BwOA Sec-8 rev 109 page 4, TS Bases 3.4.13 rev 64 page B 3.4.13-5

SRO Justification:
The question is SRO level because it requires knowledge of Tech Spec bases.

Answer Explanation: The 2PR027J is typically a leading indicator of a small SGTL. Per 2BwOA SEC-8, alternate rad monitors for identifying which SG is leaking are:
- Main Steamline rad monitors (leak not large enough to detect on these given in stem).
- N-16 rad monitors (these are portable and can be moved from to each MS line to determine which SG is leaking.)

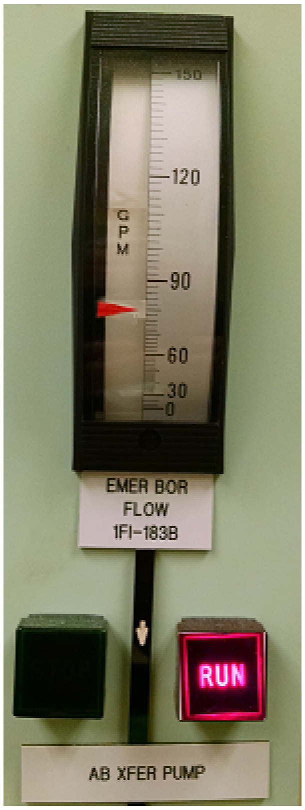
Per TS 3.4.13 bases:
The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 5). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. Leakage that exceeds this amount has been shown to have a high likelihood of degrading further to a steam generator tube rupture. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.
Distractors are variations of either unidentified or identified leakage safety analysis and/or other rad monitors that would detect SG primary to secondary tube leakage.

Choice A is incorrect, see explanation above.
Choice B is correct, see explanation above.
Choice C is incorrect; the 2PR08J rad monitor is not listed in 2BwOA SEC-8 to determine which SG is leaking as it typically monitors flow from all SG with blowdown flow aligned simultaneously. This is plausible because each steam generator has individual isolations valves.
Choice D is incorrect, see explanation above.

86 | **ID: 1267829** | **Points: 1.00**

Given:

- Unit 1 was at full power, normal alignment.
- An event occurred requiring Main Control Room evacuation.
- The crew entered 1BwOA PRI-5, CONTROL ROOM INACCESSIBILITY.
- The reactor is tripped.
- TWO control rods did NOT fully insert after the reactor trip.
- ALL other equipment operated as expected.
- The crew initiates boration from the MCR in accordance with 1BwOA PRI-5.
- When the Remote Shutdown Panel is activated, the following indications are noted. (assume the conditions are constant)
- The SRO has decided to use guidance in 1BwEP ES-0.1, REACTOR TRIP OR SI, for emergency boration.



With the above conditions, the emergency boration will first MEET the MINIMUM requirements of 1BwEP ES-0.1 for the stuck control rods in...
(assume total boration time)

A. 3 minutes.

B. 33 minutes.

C. 75 minutes.

D. 138 minutes.

Answer: B

Answer Explanation

Bwd 2016 NRC Exam Question: #86
History: New for Bwd 2016 NRC exam
SRO level
High Cog

K/A: 000068 Control Room Evacuation
A 02.02 Ability to determine and interpret the following as they apply to the Control Room Evacuation: Local boric acid flow
RO 3.7 SRO 4.2

The question meets the K/A, requires examinee to be able to determine and interpret boric acid flow from the RSP.

TIER:	1
GROUP:	2
Task No:	S-OA-34
Obj No:	T.EP01-06-C

10CFR55 Link: 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Technical Reference with Revision Number: 1BwOA PRI-5 rev108 page 3**,** 1BwEP ES-0.1 rev 204 page 5

SRO Justification:
The question is SRO level because it requires detailed knowledge of the amount of boric acid required for emergency boration. The overall mitigation strategy (RO knowledge) would be to initiate emergency boration under conditions requiring it (i.e. more than one stuck rod, natural circ condition, inadvertent RCS cooldown). knowledge of how much to borate for each given situation is detailed procedure knowledge and therefore SRO level.

Answer Explanation: 1BwOA PRI-5 directs boration to be initiated via the AB transfer pump and emergency boration valve from the BAST (vice the RWST). However, 1BwOA PRI-5 does not direct the amount of boration. Under these conditions 1BwEP ES-0.1 contains guidance of 1320 gal boric acid for each rod not fully inserted. Therefore 1320 gal x 2 = 2640 gal / 80 gpm = 33 minutes. All the distractors are emergency boration amounts for other conditions as described in 1BwEP ES-0.1.

Choice A is incorrect, amount for cooldown 2 degrees below minimum threshold of 545 degrees.
Choice B is correct, see explanation above.
Choice C is incorrect, amount for borating if on natural circulation.
Choice D is incorrect, amount for 2 stuck control rods if borating from the RWST.

87 **ID: 1267833** **Points: 1.00**

Given:

An event occurs on Unit 1 and the STA monitoring status trees notes the following indications:
- CETCs are 650ºF.
- RCS pressure is 2100 psig.

With the above conditions, the result of the Core Cooling status tree will be a(n) ____(1)_____ path and the bases for Reactor Core Safety Limits (Tech Spec 2.1.1) NOT being exceeded AT POWER is by maintaining an adequate__(2)__.

A. (1) YELLOW
(2) DNBR

B. (1) YELLOW
(2) SDM

C. (1) ORANGE
(2) DNBR

D. (1) ORANGE
(2) SDM

Answer: A

Answer Explanation
Provide Reference: 1BwST-2, pages 1 & 3 (DO NOT include page 2) **Bwd 2016 NRC Exam Question:** #87 **History:** New for Bwd 2016 NRC exam SRO level High Cog **K/A:** W/E07 Saturated Core Cooling A 02.02 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments RO 3.3 SRO 3.9 The question meets the K/A, requires examinee to adhere to status tree procedure and know bases of operations within facilities license (safety limits within tech specs). TIER: 1 GROUP: 2 Task No: S-FR-009 Obj No: T.FR02-04-C **10CFR55 Link:** 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. **Technical Reference with Revision Number:** 1BwST-2 rev 200 , TS bases 2.1.1 rev 0, page B 2.1.1-1 **SRO Justification:** The question is SRO level because it requires knowledge of Tech Spec bases as well as performance of status trees (STA/SRO function at Braidwood). **Answer Explanation:** The given parameters are plotted in the Not Acceptable region of Figure ST 2-1 therefore the subcooling is unacceptable and a yellow path is result of status trees. The bases of Tech Spec 2.1.1 reads: In MODE 1, the Departure from Nucleate Boiling Ratio (DNBR) shall be maintained $\geq$ 1.24. 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.

88	ID: 1267853	Points: 1.00

Given:

- Unit 1 is at full power, normal alignment.

The following occurs:
- Annunciator 1-7-E4, RCP THERM BARR CC WTR FLOW HIGH LOW, alarms.
- The RO verifies proper automatic actions HAVE occurred per the BwAR.
- The crew enters 1BwOA PRI-6, COMPONENT COOLING MALFUNCTION and determines that ONE RCP has a thermal barrier leak.

With the above conditions, and NO OTHER operator actions, the leak is ____(1)_____.

IF the Shift Manager then directs the crew to restore CC flow to the UNAFFECTED RCPs per 1BwOA PRI-6, the administrative requirement the SRO must complete is ____(2)_____.

A. (1) ISOLATED
(2) BwAP 1450-1, ACCESS TO CONTAINMENT, Attachment 2, Containment Entry Checklist.

B. (1) ISOLATED
(2) 1BwOL 3.7.7, LCOAR COMPONENT COOLING WATER SYSTEM

C. (1) NOT isolated
(2) 1BwOL 3.7.7, LCOAR COMPONENT COOLING WATER SYSTEM

D. (1) NOT isolated
(2) BwAP 1450-1, ACCESS TO CONTAINMENT, Attachment 2, Containment Entry Checklist

Answer: A

Answer Explanation

Bwd 2016 NRC Exam Question: #88
History: New for Bwd 2016 NRC question
SRO level
High Cog

K/A: 000026 Loss of Component Cooling Water
Gen 02.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

The question meets the K/A, requires examinee to verify the status of leak based upon MCR indications and understand that the Sm direction is going to require a containment entry.

TIER:	1
GROUP:	1
Task No:	S-AM-009
Obj No:	T.AM17-01

10CFR55 Link: 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Technical Reference with Revision Number: 1BwOA PRI-6 rev 108 page 32, BwAP 1450-1 rev 43 page 1 and 3

SRO Justification:
The question is SRO level because it requires the administrative procedure be filled out that is an SRO function.

Answer Explanation:
An automatic interlock for alarm 1-7-E4 is the closing of 1CC685 valve. This will isolate CC (containment isolation valve) flow from the RCP thermal barriers, and the line supplying CC flow has a check valve. Therefore, the leak is isolated. To restore CC to the unaffected RCPs a containment entry is required to locally isolate the affected RCP. Therefore BwAP 1450-1 is applicable. 1BwOL 3.7.7 is not applicable because the CC supply to the RCP thermal barriers is part of the CC system "service" loop (non-safety related) vice the "safety" loop (which would require LCO entry if leak developed there).

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.

89 **ID: 1267855** **Points: 1.00**

Given:

- Unit 1 was at full power, normal alignment.

The following occurs:
- The 1A SG PORV fails OPEN and CANNOT be closed.
- The crew trips the reactor and attempts to main steam isolate, but NO MSIVs will close.
- An EO locally CLOSES the 1MS019A, 1A SG PORV Block Valve.

With the above conditions, concerning the OPERABILITY status of the 1A SG PORV and the BASES for the status, the 1A SG PORV is...

A. NOT required per Tech Specs, because the SG PORV does not have a UPS.

B. INOPERABLE, because the BLOCK valve is CLOSED, ONLY.

C. INOPERABLE, because the PORV CANNOT stroke fully, ONLY.

D. INOPERABLE, because the SG PORV CANNOT stroke fully AND INOPERABLE, because the BLOCK valve is CLOSED.

Answer: C

Answer Explanation

Bwd 2016 NRC Exam Question: #89
History: New for Bwd 2016 NRC exam
SRO level
Low Cog

K/A: WE12 Uncontrolled Depressurization of all SGs
Gen 04.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.

The question meets the K/A, because it requires knowledge of the impact of locally isolating a SG PORV.

TIER:	1
GROUP:	1
Task No:	S-TS-006
Obj No:	S.MS1-15-D

10CFR55 Link: 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases.

Technical Reference with Revision Number: TS bases 3.7.4 rev 91 page B 3.7.4-3

SRO Justification:
The question is SRO level because it requires examinee knowledge of the tech spec bases.

Answer Explanation:
Per Tech Spec 3.7.4 bases "A SG PORV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand". Therefore the PORV is inoperable for not being able to stroke. Additionally 3.7.4 bases states "A closed block valve does not render it or its SG PORV line inoperable. Operator action time to open the block valve is supported in the accident analysis". This supports the correct answer that the PORV is not inoperable for reason of the block valve being closed..

Choice A is incorrect, but plausible because the 3.7.4 bases also states "To ensure that at least two SG PORVs on intact SGs are available in the event of a passive electrical failure, the uninterruptible power supply (UPS) system with at least a 90 minute battery backup supply to the C and D SG PORVs must be OPERABLE. However, this statement does not mean the PORVs without UPS supplies are not required to be operable. It is an additional requirement for the C and D PORVs.
Choice B is incorrect, see explanation above.
Choice C is correct, see explanation above.
Choice D is incorrect, see explanation above.

90	ID: 1267888	Points: 1.00

Given:

- Unit 1 was ramping up from 80% to full power.
- ALL systems were normally aligned
- The NSO was performing intermittent MANUAL rod withdrawals per the reactivity plan.

The following occurs:
- Instrument bus 113 faults and is de-energized.
- The RO notes annunciator 1-10-B5, PWR RNG FLUX HIGH ROD STOP, is in alarm.

With the above conditions, MANUAL control rod withdrawal is ___(1)____ and the procedure steps the crew will perform to clear the annunciator 1-10-B5 alarm are in ____(2)_____.

A. (1) AVAILABLE
(2) 1BwOA ELEC-2, LOSS OF INSTRUMENT BUS

B. (1) AVAILABLE
(2) 1BwOA INST-1, NUCLEAR INSTRUMENTATION MALFUNCTION

C. (1) NOT available
(2) 1BwOA ELEC-2, LOSS OF INSTRUMENT BUS

D. (1) NOT available
(2) 1BwOA INST-1, NUCLEAR INSTRUMENTATION MALFUNCTION

Answer: D

Answer Explanation

Bwd 2016 NRC Exam Question: #90
History: New for Bwd 2016 NRC exam
SRO level
Low Cog

K/A: 057 Loss of Vital AC Instrument Bus
A02.19 Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: The plant automatic actions that will occur on the loss of a vital ac electrical instrument bus.
RO 4.0
SRO 4.3

The question meets the K/A, because it requires knowledge plant interlocks resulting from a loss of an instrument bus.

TIER:	1
GROUP:	1
Task No:	S-OA-024
Obj No:	T.OA10-06

10CFR55 Link: 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Technical Reference with Revision Number: BwAR 1-10-B5 rev 51E1, 1BwOA INST-1 rev 105 page 3

SRO Justification:
The question is SRO level because it requires selection of procedure that has specific actions to bypass the alarm function that are not in the initial BwOA procedure that would not be entered initially for the malfunction.

Answer Explanation:
On a loss of instrument bus 113, PRNI N-43 will also become de-energized. This will cause a single power range High flux rod stop bistable input to alarm 1-10-B5. The coincidence for this alarm (C-2) is 1/4 and it will stop rod motion in auto and manual mode. Other rod stop alarms that will come in also are OTDT rod stop (C-3) and OPDT rod stop (C-4). These alarms will come in but the rod stop will NOT be active because the alarm coincidence (1/4) is different from the actual rod stop coincidence (2/4). This makes the AVAILABLE distractor plausible. Also there are rod stops (C-5 and C-11) that only stop auto rod motion (NOT manual).
The procedure entered for this malfunction is 1BwOA ELEC-2, however the procedural steps to clear the alarm (and regain rod withdrawl capability) are in 1BwOA INST-1. 1BwOA ELEC-2 contains a transition to BwOA INST-1 if the instrument bus is dead and cannot be re-energized.

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

91 | **ID: 1267891** | **Points: 1.00**

Given:

- Unit 1 was at full power, normal alignment.

The following occurs:
- DC bus 113 faults and is de-energized.
- The crew manually trips the reactor.
- 30 seconds later, an EO locally trips the PMG output breaker.
- All equipment operates as expected.

With the above conditions, and NO additional operator actions, one minute AFTER the reactor trip ___(1)____ MCR annunciators will be UNAVAILABLE, and after 15 minutes, ___(2)____ EAL condition is applicable.

A. (1) ALL
(2) NO

B. (1) ALL
(2) an ALERT

C. (1) NON-safety related ONLY
(2) an ALERT

D. (1) NON-safety related ONLY
(2) NO

Answer: A

Answer Explanation
Provide Reference: EP-AA-1001 Addendum 3, EMERGENCY ACTION LEVELS FOR BRAIDWOOD STATION, pages 13-36 (Hot and Cold Matrixes BW 2-1 to BW 2-24)

Bwd 2016 NRC Exam Question: #91
History: New for Bwd 2016 NRC exam
SRO level
High Cog

K/A: 058 Loss of DC Power
A02.03 Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems.
RO 3.5
SRO 3.9

The question meets the K/A, because it requires knowledge that a loss of a DC bus will have on the ability to monitor plant systems.

TIER:	1
GROUP:	1
Task No:	S-ZP-008
Obj No:	T.OA02-03

10CFR55 Link: 10CFR55.45(a)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Technical Reference with Revision Number: 1BwOA ELEC-7 rev. 002, page 3, EP-AA-1001 rev 1, addendum 3, page BW 2-5

SRO Justification:
The question is SRO level because it requires assessment of plant conditions for an EAL determination.

Answer Explanation:
On a loss of Non-ESF DC 113 bus, and a reactor trip, power will be loss to AC busses 143 and 157 also. Bus 143 (via MCC 133V2) and DC bus 113 are the normal and reserve power supplies to the 1PA19J, Annunciator Logic Cabinet. With both power supplies lost, NO annunciators will be available as they are processed thru the logic cabinet. If the DC bus lost had been DC bus 114, the result would have been a similar loss of complete power to the 1PA30J (Train N) which would result in just losing NON-safety related annunciator sections.
A recent revision to the Braidwood EAL matrix (rev. 6), changes the loss of all annunciators (combined with a reactor trip) from an Alert classification to No classification. The EAL threshold changed from a "loss of annunciators" to a "loss of MCR indications". With the new EALs, as long as the operator can monitor major parameters (from MCR indications, meters, etc.) then no EAL classification is applicable.
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.

92 **ID: 1267908** **Points: 1.00**

Given:

- Unit 1 is at 50% power.

Subsequently:

- 1IA066, IA Inside CNMT Isol Valve, fails and slowly closes.
- The crew enters 1BwOA SEC-4, LOSS OF INSTRUMENT AIR.
- 1CV121, CV Pump Flow Control Valve, controller was taken to manual and throttled to reduce charging flow to 40 gpm.
- PZR pressure is 2275 psig and slowly rising.
- PZR level is 45% and slowly rising.
- VCT level is 50% and lowering.

Given the above conditions, per 1BwOA SEC-4, which of the following would require an IMMEDIATE reactor trip?

A. Pressurizer level rises to 80%.

B. The operating CV pump is tripped.

C. Pressurizer pressure rises to 2350 psig.

D. Volume Control Tank level drops to 37%.

Answer: C

Answer Explanation

Bwd 2016 NRC Exam Question: #92
History: Bank from 2014 Bwd NRC exam
SRO level
High Cog

K/A: 065 Loss of Instrument Air
Gen 2.4.11 Knowledge of abnormal condition procedures.
RO 4.0 SRO 4.2

The question meets the K/A, because it requires detailed knowledge of 1BwOA SEC-4.

TIER:	1
GROUP:	1
Task No:	S-OA-103
Obj No:	T.OA39-05

10CFR55 Link: 10CFR55.45(a)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Technical Reference with Revision Number:
1BwOA SEC-4 rev 105, page 13.

SRO Justification:
The question is SRO level because it requires detailed knowledge of 1BwOA SEC-4 beyond the overall mitigation strategy.

Answer Explanation:
Since PZR pressure is 2275 psig and rising and with no instrument air available, PZR sprays will not be available to lower pressure. This means that PZR pressure will be maintained by the PZR PORVs. Per 1BwOA SEC-4 if you are unable to maintain PZR pressure less than 2335# using PZR heaters or sprays it directs the crew trip the reactor.

A is incorrect. If PZR level reaches 80%, 1BwOA SEC-4 directs the CV pump to be tripped but not the reactor to be tripped.
B is incorrect. As long as CC flow is maintained to the RCPs and RCP temperatures are monitored to not exceed limits, tripping the reactor is not required when no CV pump is running.
C is correct. See explanation.
D is incorrect. If VCT level drops below 10%, 1BwOA SEC-4 directs the crew to monitor RMCS makeup and swap CV pump suction to the RWST if RMCS makeup is not adequate.

93	ID: 1267909	Points: 1.00

Given:

- A 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 is 6 psig and rising.
- CETCs indicate 720°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
Bwd 2016 NRC Exam Question: #93 **History:** Bank from 2009 Bwd NRC exam SRO level High Cog **K/A:** W/E05 Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink) A02.02 Adherence to appropriate procedures and operation within the limitations in the facility*s license and amendments. RO 3.7 SRO 4.3 The question meets the K/A, because it requires proper emergency procedure adherence. TIER: 1 GROUP: 1 Task No: S-EP-005 Obj No: T.FR03-03 **10CFR55 Link:** 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. **Technical Reference with Revision Number:** 1BwEP-0 rev 207 page 12 **SRO Justification:** The question is SRO level because it requires assessment of conditions and selection of appropriate procedure. In this condition the procedure transition is prior to status tree monitoring and not to a major EOP. **Answer Explanation:** Choice A is incorrect, SG levels are not currently adequate with adverse containment. Requirement is 31% or greater. This would be the transition otherwise that is why it is plausible. Choice B is incorrect, status tree monitoring does not take effect 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.

94	ID: 1267911	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
Bwd 2016 NRC Exam Question: #94 **History:** Bank from 2009 Bwd NRC exam SRO level High Cog **K/A:** Gen: Conduct of Operations Gen 2.1.34 Knowledge of primary and secondary plant chemistry limits RO 2.7 SRO 3.5 The question meets the K/A because the candidate have knowledge of chemistry limits and when they are required to be sampled for to meet Tech Spec surveillance requirements. TIER: 3 GROUP: 1 Task No: S-TS-008 Obj No: 8E.TS-008 **10CFR55 Link:** 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases. **Technical Reference with Revision Number:** Tech Spec 3.4.16 rev 165 page 3.4.16-2 **SRO Justification:** The question is SRO level because it is the SROs responsibility to ensure the surveillance frequency is met by contacting the chemistry department following a turbine ramp of $\geq$ 15% RTP within a 1 hour period. **Answer Explanation:** 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.

95	ID: 1267912	Points: 1.00

Given below are 4 different evolutions of MOVING spent fuel:

(1) Moving RECENTLY Irradiated Spent Fuel in the REACTOR VESSEL.

(2) Moving RECENTLY Irradiated Spent Fuel in the SPENT FUEL POOL.

(3) Moving IRRADIATED Spent Fuel in the REACTOR VESSEL.

(4) Moving IRRADIATED Spent Fuel in the SPENT FUEL POOL.

Which of the following list the evolutions that REQUIRE a Licensed Supervisor with an active SRO License to be present?

A. 1 ONLY

B. 1 & 2 ONLY

C. 1 & 3 ONLY

D. 1, 2, 3 & 4

Answer: C

Answer Explanation

Bwd 2016 NRC Exam Question: #95
History: Modified from 2014 Bwd NRC exam (SG1027-N14-94)
SRO level
Low Cog

K/A: Gen: Conduct of Operations
2.1.36 Knowledge of procedures and limitations involved in core alterations.
RO 3.0 SRO 4.1

The question meets the K/A because the examinee must have knowledge of procedures and limitations involved in core alterations.

TIER:	3
GROUP:	1
Task No:	S-FH-005
Obj No:	T.GP06-05

10CFR55 Link: 10CFR55.43(b)(6) Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.

Technical Reference with Revision Number: OU-AP-200 rev 20 page 6

SRO Justification:
SRO level because the SRO is the one responsible for having an understanding of what procedure requirements and limitations are required with core alterations, and also when a core alteration is occurring.

Answer Explanation: Per OU-AP-200 the Licensed Supervisor with an active SRO License must be present at the refuel cavity with no other concurrent responsibilities whenever a core alteration is in progress. Core Alts are defined in Tech Specs as: CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The distractors of movements within the pool do not require an SRO licensed supervisor (however, it is allowed) and can be directed by a non-licensed supervisor. RECENTLY irradiated vs. Irradiated fuel is also a distractor that is defined in Tech Spec and applies to several Tech Spec applicability statements and required actions.

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.

96 **ID: 1267929** **Points: 1.00**

Given:

- Unit 1 is at full power, normal alignment.

The following sequence of events then occur on 1/1/16:

- At 1200: Bus 141 is declared INOPERABLE for an emergent maintenance issue.
- LCO 3.8.9, Distribution Systems, Cond. A is entered with an 8 hour completion time to restore the bus to operable.

- At 1800: The rounds EO discovers an UNISOLABLE SX leak (ASME code piping thru wall leak) in the 1A SI pump room. NO other equipment is affected by the leak.
- LCO 3.7.8, SX System, Cond. A is entered with an 72 hour completion time to repair SX to the 1A SI pump.

- At 1900: Bus 141 repairs are complete and the bus is declared operable.
- LCO 3.8.9 is exited.

Currently the SRO is evaluating 1BwOL 3.5.2, LOCAR ECCS-OPERATING, entry for the 1A SI pump.

With the above conditions, LCO 3.5.2 entry is REQUIRED NO later than...

A. 1800 on 1/4/16.

B. 1900 on 1/4/16.

C. 1200 on 1/8/16.

D. 1800 on 1/15/16.

Answer: A

Answer Explanation
Provide Reference: 1BwOL 3.5.2 **Bwd 2016 NRC Exam Question:** #96 **History:** New for Bwd 2016 NRC exam SRO level High Cog **K/A:** Gen: Equipment Control 2.2.21 Knowledge of pre- and post-maintenance operability requirements. RO 2.9 SRO 4.1 The question meets the K/A because the examinee must have knowledge of how to apply operability requirements. TIER: 3 GROUP: 2 Task No: S-TS-006 Obj No: S.TS1-06-A **10CFR55 Link:** 10CFR55.43(b)(2) Facility operating limitations in the technical specifications and their bases. **Technical Reference with Revision Number:** 1BwOL 3.5.2 rev 009, TS 3.0.6 rev 98 page 3.0-3 **SRO Justification:** SRO level because the SRO is the one responsible for evaluating and implementing delayed LCO entries. **Answer Explanation:** Per LCO 3.0.6, Rule 2: In the event additional SUPPORT SYSTEM(s) become inoperable during the Completion Time for restoration of the first SUPPORT SYSTEM, the LCOAR entry(s) of the SUPPORTED SYSTEM may be delayed by either the maximum allowed Completion Time of the SUPPORT SYSTEMs, or 2 times the Completion Time for restoration of the SUPPORTED SYSTEM (applied at the time the second SUPPORT SYSTEM becomes inoperable), whichever is less. This is applied at the time the second support system is declared inoperable. 1BwOL 3.5.2 is supplied because the intent is not to know the rule from memory, but rather be able to apply the rule for a given situation. Also the candidate needs to know LCO 3.5.2, condition A completion time of 7 days (not expected from memory). Choice A is correct, see explanation above. Choice B is incorrect, This is the rule applied from the time bus 141 was restored (vice SX leak isolated). Choice C is correct, This is seven days from the first support system (bus 141 inoperability). Choice D is incorrect, This is the rule applied with the longer of the two options (vice the shorter option).

97 **ID: 1267932** **Points: 1.00**

Given:

- Unit 1 Main Turbine was just runback from 100% power due to a secondary pump trip.
- The crew is performing 1BwOA SEC-1, SECONDARY PUMP TRIP.

- Annunciator 1-10-A6, ROD BANK LO-2 INSERTION LIMIT, is LIT.
- Annunciator 1-10-B6, ROD BANK LOW INSERTION LIMIT, is LIT.

- 1BwOSR 3.1.1.1-2, SDM SURVEILLANCE DURING OPERATIONS, resulted in 1200 pcm available shutdown reactivity.

Note: procedure names given for reference.

- 1BwOA PRI-2, EMERGENCY BORATION
- BwOP CV-6, OPERATION OF THE REACTOR MAKEUP SYSTEM IN THE BORATE MODE/BATCH BORATION METHOD

With the above conditions, in accordance with 1BwOA SEC-1, the US will direct the crew to borate the RCS per...

A. 1BwOA PRI-2, and boration is required by Tech Specs.

B. 1BwOA PRI-2, but boration is NOT required by Tech Specs.

C. BwOP CV-6, but boration is NOT required by Tech Specs.

D. BwOP CV-6, and boration is required by Tech Specs.

Answer: D

Answer Explanation
Bwd 2016 NRC Exam Question #97 **History:** Modified from Bwd 2011 NRC exam (SS20045-N01) SRO level Low Cog **K/A:** Gen: Equipment Control 2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems. RO 3.9 SRO 4.5 The question meets the K/A because the candidate must know 1 hr or less action statement in tech specs. TIER: 3 GROUP: 2 Task No: S-OA-096 Obj No: T.OA36-03 **10CFR55 Link:** 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. **Technical Reference with Revision Number:** 1BwOA SEC-1 rev. 106 page 9, TS 3.1.6 rev. 98 page 3.1.6-1 **SRO Justification:** The question is SRO level because it requires assessment of facility conditions and selection of proper procedure. **Answer Explanation:** With control rods below the rod insertion limit (RIL) LOW alarm and below the RIL LO-2 alarm, 1BwOA SEC-1 requires borating the RCS and Tech Specs require either SDM be adequate or boration. The COLR minimum requirement for SDM is 1300 pcm. Therefore, at 1200 pcm calculated SDM, boration is required. However, neither situation is 1BwOA PRI-2 entry criteria for emergency boration. A is incorrect, 1BwOA PRI-2 entry conditions not met. B is incorrect, 1BwOA PRI-2 entry conditions not met. C is incorrect, boration is required by tech specs. D is correct, see explanation above.

98 **ID: 1267933** **Points: 1.00**

Given:

- A Unit 2 containment release is pending.
- The SRO is filling out RP-BR-980, CONTAINMENT RELEASE FORM and evaluating whether to approve the release with the current exhaust fan configuration.

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 SRO to NOT APPROVE 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

Bwd 2016 NRC Exam Question: #98
History: Bank from Bwd 2009 NRC exam
SRO level
High Cog

K/A: Gen Radiation Control
2.03.06 Ability to Approve Release Permits
RO 2.0 SRO 3.8

The question meets the K/A because the candidate must know when conditions for a release are/are not met.

TIER:	3
GROUP:	3
Task No:	S-HP-002
Obj No:	S.VP1-09-B

10CFR55 Link: 10CFR55.41(b)(13) Procedures and equipment available for handling and disposal of radioactive materials and effluents

Technical Reference with Revision Number: RP-BR-980 rev 18, page 18

SRO Justification:
The question is SRO level because the answer information is contained within the procedure section that is specifically identified as "SRO Responsibility".

Answer Explanation: RP-BR-980 section D reads as follows:
D. SHIFT ENGINEER OR SRO RESPONSIBILITY
1. PLACE the placard "Gaseous Release in Progress" somewhere on 0PM02J where the placard will be visible. This is to ensure that for a unit 1 release an Aux. Bldg. Exhaust Fan (unit 1 side) or a Lab HVAC exhaust fan is in operation, OR for a unit 2 release an Aux. Bldg. Exhaust Fan (unit 2 side) or a Radwaste Bldg. Exhaust Fan is in operation, **or** for either unit , to have (2) Charcoal Booster Fans (0VA03CA-F) and (1) FHB Charcoal Booster Fan (0VA04CA or 0VA04CB) in operation.

When a containment release is in progress, RP-BR-980 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 (with no VA exh fan running)
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.

99 **ID: 1269457** **Points: 1.00**

Given:

- An RCS LOCA occurred on Unit 1.
- The crew has implemented 1BwCA-1.2, LOCA OUTSIDE CONTAINMENT.
- Containment Floor Water Level is 0 inches.
- RWST level is 50% and slowly lowering.
- After completing the actions of 1BwCA-1.2, RCS pressure is still lowering.

Based on the above conditions, the NEXT procedure the Unit Supervisor will implement is…

A. 1BwCA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

B. 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

C. 1BwEP ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.

D. 1BwEP ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

Answer: A

Answer Explanation
Bwd 2016 NRC Exam Question: #99 **History:** Bank from Bwd 2009 NRC exam SRO level High Cog **K/A:** Gen Emergency Procedures/Plan 2.4.04 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures RO 4.5 SRO 4.7 The question meets the K/A because the candidate must be able to recognize entry conditions for an EOP. TIER: 3 GROUP: 4 Task No: S-CA-011 Obj No: T.CA2-02 **10CFR55 Link:** 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. **Technical Reference with Revision Number:** 1BwCA-1.2 rev 200 page 7 step 3.a **SRO Justification:** The question is SRO level because it requires assessment of plant conditions and selection of emergency procedure. **Answer Explanation:** After completing actions of 1BwCA-1.2, RCS pressure is lowering and leak is unisolable. 1BwCA-1.2 directs transition to 1BwCA-1.1 if an unisolable leak is present upon completing actions of 1BwCA-1.2. Transition from 1BwCA-1.2 is entry condition for 1BwCA-1.1. A is correct, see explanation above. B is incorrect, transition to 1BwEP-1 is directed if leak is isolated. C is incorrect, Not at RWST lo-2 setpoint yet. Transition is directed to either 1BwEP-1 or 1BwCA-1.1, depending if leak is isolated. D is incorrect, transition is directed to 1BwCA-1.1, this is plausible becasue the strategy used in 1BwEP ES-1.2 is similiar to the strategy used in 1BwCA-1.1, however 1BwEP ES-1.2 does not account for the depletion of the RWST.

100	ID: 1267953	Points: 1.00

Given:

- A security event has been declared at Braidwood Station.
- The crew has entered 0BwOA SECURITY-1, SECURITY THREAT.
- The event has resulted in an explosion with a breach of containment integrity.

Which of the following documents will be used to provide a framework for the Operations and Security Personnel interface?

A. Extensive Damage Mitigation Guidelines (EDMG)

B. Operational Contingency Action Guideline (OCAG)

C. Severe Accident Management Guidelines (SAMG)

D. Offsite Dose Calculation Manual (ODCM)

Answer: B

Answer Explanation
Bwd 2016 NRC Exam Question: #100 **History:** New for Bwd 2016 NRC exam SRO level Low Cog **K/A:** Gen: Emergency Procedures/Plan 2.4.28 Knowledge of procedures relating to a security event (non-safeguards Information) RO 3.2 SRO 4.1 The question meets the K/A because the candidate must have knowledge of which guideline is used for the purpose of coordinating activities between operations and security during a security event. TIER: 3 GROUP: 4 Task No: S-OA-143 Obj No: 7D.OA-060-B **10CFR55 Link:** 10CFR55.43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. **Technical Reference with Revision Number:** 0BwOA SECURITY-1 rev 17, page 3 **SRO Justification:** The question is SRO level because it requires knowledge of guidelines that are used only by the SROs. **Answer Explanation:** The OCAGs are the correct answer (see below). The other guidelines are all documents that may be referred to during a security event with extensive damage, however the OCAGs are the only one referred to in 0BwOA SECURITY-1 to refer to initially. Choice A is incorrect, The EDMG is to provide Initial operator action to respond to a large fire/explosion causing extensive plant damage. Choice B is correct, The purpose of OCAG is to provide a framework for the Shift Manager and Security Personnel interface. Choice C is incorrect, The SAMG is to provide actions for guidance for fast-acting core damage sequences. Choice D is incorrect, The ODCM is intended for the calculation of radiation doses during routine (i.e non-accident conditions).