PROPOSED WRITTEN

RO QUESTIONS 1-75

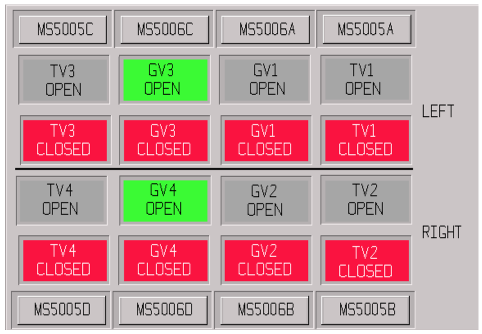
SRO QUESTIONS 76-100

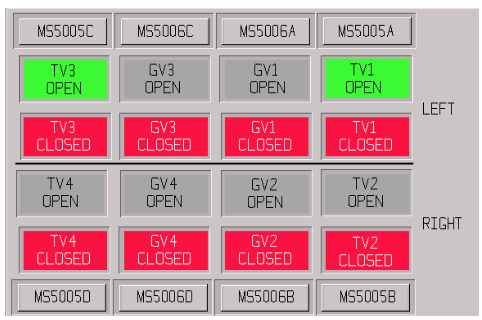
QUESTION 001 **Points: 1.00**

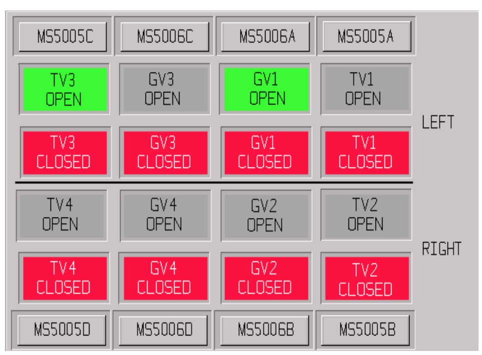
Given:

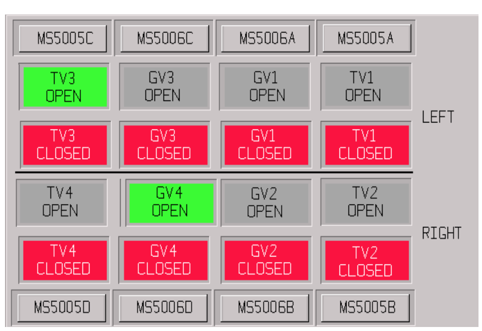
- Unit 1 was in normal at power alignment.
- A reactor trip/turbine trip occurred.
- NO operator actions have been taken.

Which of the following screen shots from DEHC Graphic Panel 5501 indicate that Main Steam flow to the HP turbine is occuring?

a.


b.


c.


d.


Answer:
c.

Explanation:

The question meets the K/A by requiring the examinee to verify whether a turbine trip has occurred following a reactor trip. In order for main steam to be completely isolated from the HP turbine either both TVs or both GVs must be closed on both sides (left and right) of the HP turbine.

Choice a. is incorrect, both sides of the turbine have both TVs closed.

Choice b. is incorrect, both sides of the turbine have both GVs closed.

Choice c. is correct, a throttle valve and governor valve, both on the left side of the turbine are in the intermediate position.

Choice d. is incorrect, the LH side of the turbine has both GVs closed and the RH side has both TVs closed.

2 **Points: 1.00**

Given:

- Unit 1 was at full power, normal alignment, MOL, for the past nine months.
- A small break RCS LOCA occurred.
- A reactor trip and SI were initiated and all equipment operated as designed.
- Decay heat is maintaining RCS Tave at 557°F.
- The crew is currently performing 1BwEP ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.
- Prior to starting an RCS cooldown, 1BwOSR 3.1.1.1-1, SHUTDOWN MARGIN DAILY VERIFICATION DURING SHUTDOWN, is initiated.
- Initial post trip shutdown margin has been verified adequate per 1BwOSR 3.1.1.1-1.

DURING the RAPID cooldown, 1BwOSR 3.1.1.1-1 will ensure adequate shutdown margin by verifying...

a. adequate RCS boron concentration.

b. RCS cooldown rate is maintained within limits.

c. all control rods remain at the core bottom position.

d. core xenon levels are accounted for.

Answer:
a.

Explanation:

The question meets the K/A, requires examinee knowledge of surveillance procedures used during a small break LOCA.
1BwOSR 3.1.1.1-1 initially verifies the first 12 hours of shutdown margin (SDM) by normal post reactor trip responses (all rods at bottom, no initial RCS dilution and RCS temp above 550°F. Following the initial SDM verification, SDM is verified by verifying/borating to cold shutdown conditions.
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, rapid cooldown section of surv. does not include xenon calculations.

3 **Points: 1.00**

During a recovery from a Large Break LOCA, the LOSS of which of the following will have the GREATEST impact on LONG TERM CORE COOLING?

a. Safety Injection Pumps.

b. Centrifugal Charging Pumps.

c. Residual Heat Removal Pumps.

d. Reactor Coolant Pumps.

Answer:
c.

Explanation:
The question meets the K/A, requires examinee knowledge of interrelation between a large break LOCA and ECCS pumps.
Long term core cooling following a large break LOCA is supported by the recirc phase of ECCS. The RH pumps are the source of cooling water during the recirculation phase of an RCS LOCA by having their suction aligned to the containment recirc sump. They provide the largest volume of water from ECCS system during long term cooling as well as supplying the suction to the SI and CV pumps during the recirc phase.
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, RCPs will have little effect during a large break LOCA since RCS loops will be essentially voided and RCP starting requirements need RCS pressure at approx. 240 psig.

4 **Points: 1.00**

Given:

- Unit 1 RCPs are being started following a refueling outage.
- Currently the 1D RCP is running.
- All the other RCPs are available for a start with the following parameters:

	1A RCP	1B RCP	1C RCP
Seal Injection Flow	10 gpm	8 gpm	11 gpm
Seal Leakoff Flow	0.3 gpm	0.5 gpm	0.6 gpm
#1 Seal DP	360 psig	355 psig	350 psig
SEAL LEAKOFF FLOW LOW alarm	LIT	LIT	LIT

With the above conditions, which (if any) RCP should be started NEXT and why?

a. 1A RCP because it has the lowest seal leakoff flowrate.

b. 1B RCP because it has the lowest seal injection flowrate.

c. 1C RCP because it will provide pressurizer spray flow.

d. None, because the SEAL LEAKOFF FLOW LOW alarm must be cleared prior to starting an RCP.

Answer:
a.

Explanation:

The question meets the K/A, requires examinee ability to apply and explain system limits and precautions. Braidwood has had several events of low RCP seal leakoff flow on idle RCPs when returning to service from an outage. The precaution described below is a result of this OPEX.

BwOP RC-1, precaution D.4 directs starting the _D RCP first, then the RCP with the lowest seal return flow. Minimum seal return flow to allow starting an RCP is 0.2 gpm. Seal leakoff flow will rise as the RCP is started. Because all RCP seal leakoffs join together into a common header to the VCT, as each RCP is started and its seal leakoff flow rises, more back pressure is applied to the remaining RCP seal leakoff lines. Therefore, the RCP with the lowest seal leakoff flow is started first (after pressurizer spray is established) to prevent the rise in back pressure from reducing seal leakoff flow below the 0.2 gpm minimum on an idle pump.

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, seal leakoff low alarm setpoint is 0.82 gpm, however minimum leakoff flow for starting an RCP is 0.2 gpm.

5 **Points: 1.00**

Given:

- Unit 1 was at full power, normal alignment.
- The 1A CV pump was operating normally with the following indications:
  - Motor Amps - 65
  - Pump Disch. Press. - 2445 psig
  - 1FI-121A CHR HDR FLOW - 134 gpm
  - 1PI-120A CHR HDR PRESS - 2342 psig

Subsequently, annunciator 1-9-D3, CHG LINE FLOW HI LOW alarms.

- Currently, the NSO notes the 1A CV pump indications have stabilized as follows:
  - Motor Amps - 50
  - Pump Disch. Press. - 2610 psig
  - 1FI-121A CHR HDR FLOW - 0 gpm
  - 1PI-120A CHR HDR PRESS - 2250 psig

Which of the following events could have caused the 1A CV pump indications to change as described above?

a. The CV pump shaft has broken causing the impeller to stop rotating.

b. The manual discharge valve stem separated from the disc and the disc fell into the flow stream.

c. A discharge pipe break has resulted in the pump operating in runout condition.

d. A VCT level control problem has resulted in the pump operating in a cavitation condition.

Answer:
b.

Explanation:

The question meets the K/A, requires examinee ability to determine and interpret charging pump problems. A separated discharge valve disc (same effect as closing the discharge valve) would cause the given indications. Pump flow would drop to approx. 60 gpm through the recirc flow path.
Amps would drop (less flow)
Pump disch. press. would rise (more back pressure)
CHR HDR FLOW would drop to 0 (measured downstream of disch valve)
CHR HDR PRESS would drop to approx. RCS pressure.
Choice A is incorrect, a shaft break would cause pump discharge pressure to drop to 0.
Choice B is correct see explanation above.
Choice C is incorrect, runout condition would raise amps and drop disch. pressure.
Choice D is incorrect, cavitation would cause fluctuating amps and disch. pressure.

6 **Points: 1.00**

Given:

- Unit 2 reactor is at full power.
- The crew placed excess letdown in service one hour ago in conjunction with normal letdown.
- 2CV8143, Excess Letdown to Seal Filter or RCDT Valve, is in the VCT position.

The following indications have been noted by the crew:

- VCT level has been slowly lowering for the past 15 minutes.
- ALL RCP seal injection flows have been stable for the past 55 minutes.
- ALL RCP seal return flows have been stable for the past 55 minutes.
- CC surge tank level is currently 76% and has been rising slowly for the past 15 minutes.
- 2PR09J, Unit 2 CC Heat Exchanger Rad Monitor, is in an ALERT condition.

Based on the above conditions, a leak has developed in which of the following heat exchangers?

a. Seal Water

b. 2B RCP Thermal Barrier

c. Excess Letdown

d. Letdown

Answer:
d.

Explanation:

The question meets the K/A, requires examinee ability to determine the location of a leak in the CC system. With CC surge tank rising and 2PR09J in alert condition, the leak is IN TO the CC system (vs. OUT OF). Therefore the system pressure of the leaking fluid must be higher than CC system pressure.
CC system pressure ~140 psig.

Choice A is incorrect, Seal Water HX ~15-25 psig (a little higher than VCT Pressure)
Choice B is incorrect, RCP thermal barrier is above RCS pressure (2250 psig), however the seal injection flows are stable which eliminates this as a possible source.
Choice C is incorrect, Excess L/D is at RCS pressure (2235 psig), however the seal return flows are stable which eliminates this as a possible source.
Choice D is correct, L/D HX pressure ~ 400 psig

7 **Points: 1.00**

Given:
- Unit 1 is at 100% power, normal alignment.
- 1PK-455A, MASTER PZR PRESS CONT output signal has failed high in AUTO and MAN.

With the above condition, the RO will manually adjust 1PK-455B and 1PK-455C, SPRAY VALVE CONTROLLERS to...

a. prevent opening of the pressurizer safety valves.

b. prevent the RCS from reaching DNB condition.

c. allow auto reclosure of the pressurizer PORVs.

d. prevent lifting of the pressurizer PORVs.

Answer:
b.

Explanation:
The question meets the K/A, requires examinee knowledge of reason for taking manual control of pressurizer spray valves and closing them.
Failing master pressure controller output signal high will de-energize all pressurizer heaters and open pressurizer spray valves resulting in RCS pressure lowering.
Choice A is incorrect, pressurizer safety valves will not open on lowering pressure.
Choice B is correct, lowering pressurizer pressure will result in RCS DNB condition.
Choice C is incorrect, since PORVs will not open, they will not have to be reclosed.
Choice D is incorrect, pressurizer PORVs operate on high pressure bistables and will not open on lowering pressure.

8 **Points: 1.00**

Given:

- Unit 1 has experienced an ATWS condition and the crew is performing 1BwFR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS.
- An operator is dispatched to locally trip the reactor from the MEER.
- BOTH U-1 Reactor Trip Breakers FAIL to open locally.
- Subsequently, the operator performs the following actions in the listed order:

1. OPENS the 1A MG SET GENERATOR Breaker.
2. OPENS the 1A MG SET MOTOR Breaker.
3. OPENS the 1B MG SET GENERATOR Breaker.
4. OPENS the 1B MG SET MOTOR Breaker.

With the above conditions, the reactor tripped WHEN the operator performed action...

a. 1

b. 2

c. 3

d. 4

Answer:
c.

Explanation:
The question meets the K/A, requires examinee knowledge of the interrelationship between breakers in the rod drive power system and an ATWS condition.
1BwFR S-1 step 6 RNO has the operator locally perform actions "until the reactor is tripped". The MG sets providing power to the reactor trip breakers are arranged in parallel and then crosstied at the generator output before feeding in series trip breakers. Either MG set provides enough power to maintain the rod control system energized. Therefore, power will not be interrupted to the rod drive power cabinets until at least one generator or one motor breaker is open from both trains of MG sets.
Choice A is incorrect, power is still available from the 1B MG set.
Choice B is incorrect, power is still available from the 1B MG set.
Choice C is correct, see explanation above.
Choice D is incorrect, reactor trip occurred at previous step.

9 **Points: 1.00**

Given:
- Unit 2 is at full power, normal alignment.
- An instantaneous 600 gpm steam generator tube rupture occurs.
- NO operator action is taken as the crew attempts to diagnose the event.

With the above conditions, pressurizer PRESSURE will initially drop and then during the next two (2) minutes...
(assume no automatic reactor trip setpoint is reached)

a. continue to drop, IN direct proportion to the tube rupture leak rate.

b. recover due to the make up capability of the automatic level control system.

c. continue to drop, but NOT in direct proportion to the tube rupture leak rate.

d. recover due to the capability of the automatic pressure control system.

Answer:
c.

Explanation:
The question meets the K/A, requires examinee to have operational knowledge of leak rate vs. pressure drop during a SGTR.
A 600 gpm tube rupture represents the approx. leak rate of a design basis rupture (full "shear" of 1 ruptured tube at normal operating pressures. Since the pressurizer is operating under saturation conditions, ideal gas laws are not applicable (i.e pressure will not drop directly proportional to leak rate). Several other factors will combine to slow (but not stop) the pressure loss over time. Among these factors are, where the pressurizer atmosphere is in relation to the saturation curve, the relative DP between the primary and secondary pressures, the rise in charging flow until maximum rate is achieved, the rise in pressurizer heater demand until maximized. A combination of all these variables will work to slow the pressure drop, thus making it NOT directly proportional to the actual leak rate.
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.

10 **Points: 1.00**

Given:

- Unit 2 was at 100% power.
- All systems were normally aligned.
- A loss of DC Bus 212 occurs and the plant responds as expected.
- The crew has implemented 2BwEP-0, REACTOR TRIP OR SAFETY INJECTION, and has transitioned to 2BwEP ES-0.1, REACTOR TRIP RESPONSE.
- Currently, ALL SG narrow range levels are 25% and slowly lowering.

With the above conditions and assuming NO operator actions on the AFW system, which of the following is the current Auxiliary Feedwater system status?

	2A AF Pump	"A" Trn FCVs 2AF005A-D	2B AF Pump	"B" Trn FCVs 2AF005E-H
a.	RUN	THROTTLED	STOP	CLOSED
b.	RUN	THROTTLED	STOP	FULL OPEN
c.	STOP	FULL OPEN	STOP	FULL OPEN
d.	RUN	THROTTLED	RUN	THROTTLED

Answer:
b.

Explanation:

Question meets KA - question requires examinee ability to determine proper operation of AF pumps and regulating valves following a loss of main feed water.

A loss of DC bus 212 will cause all the feedwater isolation valves to fail close, thus requiring a reactor trip. Post reactor trip plant response from 100% power will cause steam generator levels to drop below the lo-2 setpoint (36%) and cause an automatic AF actuation. With a loss of DC bus 212, the 2A AF pump will start but the 2B will not. The A trn FCVs will automatically throttle to control AF flow at approx. 170 gpm. The B train FCV controllers will be also be set to throttle flow, however with no pump running, the FCVs will be full open (stand by position) since 170 gpm is being demanded but 0 gpm is being sensed.

A is incorrect, would be correct for a loss of instrument bus that affected B train FCVs after flow is sensed.

B is correct, see explanation above.

C is incorrect, would be correct for same situation on unit 1 (AF setpoint is 18% on unit 1).

D is incorrect, would be correct for a normal post reactor trip response.

11 **Points: 1.00**

Given:

- BOTH UNITS have experienced a severe electrical transient and have lost ALL AC power.
- BOTH UNITS are performing 1/2BwCA 0.0, LOSS OF ALL AC POWER.
- BOTH UNITS have indications of an RCS LOCA developing.
- Local operators are being dispatched to shed DC power loads from the 125 VDC class 1E busses.

With the aove conditions, why is it necessary to shed loads from the DC busses? It is necessary because the design basis for the 125 VDC class 1E system is...

a. ONLY for a safe shutdown from full power combined with a loss of OFFSITE power.

b. ONLY for a safe shutdown from full power combined with a loss of ALL AC power.

c. for a design basis LOCA combined with a loss of ALL AC power, BUT non-class 1E loads must be shed.

d. ONLY for a design basis LOCA combined with a loss of OFFSITE power.

Answer:
d.

Explanation:
The question meets the K/A, requires examinee knowledge of reasons for length of time battery capacity is designed.
Per the UFSAR and 125 VDC ESF lesson plan, the 125 VDC class 1E battery systems are designed for a DBA (LOCA) combined with a loss of off site power. The only specific time mentioned is the battery rating in the UFSAR of 2320 amp -hours at the 8 hour rate to an end voltage of 1.75 volts per cell. However, the actual time which a battery would last is dependent upon load. The UFSAR also mentions that the batteries are designed for a 1 hour design duty cycle in the event of a LOCA concurrent with a loss of off site power. A loss of ALL AC is not a design basis event (thus a contingency action procedure was developed to address). The load shedding directed in an extended loss of all AC power is to maximize the battery life for monitoring MCR indications. But the batteries are not designed specifically for an extended loss of all AC power. The important concept for the operator to know is not the specific amount of time the battery was designed to last, but the event that the battery is designed to protect for.
Choice A is incorrect, the design basis includes a DBA (LOCA), not only a safe shutdown from 100%.
Choice B is incorrect, loss of ALL AC is not in design basis of plant or DC system.
Choice C is incorrect, loss of ALL AC is not in design basis of plant or DC system even if non-ESF loads are shed.
Choice D is correct, see explanation above.

12 **Points: 1.00**

Given:

- Unit 2 is at 13% reactor power, with a unit start up in progress.
- U-2 Main Generator preps for synchronization are in progress.
- Unit 1 is at 100% power, normally aligned.

The following then occurs:

- A Unit 2 SAT fault results in a loss of off site power on Unit 2.
- All plant systems operate as designed.
- The crew is currently performing the actions to crosstie Non-ESF buses to ESF buses per 2BwEP ES-0.1, REACTOR TRIP RESPONSE.

With the above conditions, for Unit 2, the most time critical reason for re-energizing a Non-ESF bus is...

a. prevent over-heating of WS cooled Non-ESF equipment.

b. re-establishing positive RCS pressure control.

c. re-establishing positive RCS inventory control.

d. preventing Main Generator hydrogen from escaping.

Answer:
b.

Explanation:

The question meets the K/A, requires examinee knowledge of reason for actions contained in EOP for loss of off site power.

The background document for 2BwEP ES-0.1 gives examples of non-ESF loads that may need to be repowered for long term recovery as air compressor and pressurizer heaters. Operating experience (INPO OE12279) has shown that the loss of pressurizer heaters can lead to lowering of RCS pressure and a low pressurizer pressure SI. Because of this, the restoration steps of 2BwEP ES-0.1 are prioritized with restoring pressurizer heaters first.

Choice A is incorrect, WS will be available from the WS pumps supplied by Unit 1 (0A and 0B) which have low pressure auto start interlocks if necessary.

Choice B is correct, see explanation above.

Choice C is incorrect, instrument and station air will be available from the SACs supplied by Unit 1 (U1 and U0) which have low pressure auto start interlocks if necessary. Therefore, RCS inventory control from charging and letdown will remain available.

Choice D is incorrect, although power will be lost to the air side and hydrogen side seal oil pumps and loss of generator hydrogen has the potential to be a serious plant problem, the seal oil system will remain in operation via the back up oil supplied from the bearing oil pump which is powered from ESF bus 131X.

13 **Points: 1.00**

Given:

- Both units were at 100% power, normal alignment.
- Subsequently, 4KV bus 142 faults and its feed breaker trips.
- Ten minutes later, a dispatched local operator reports the following indications at DC Bus 112:
  - Battery Current - 50 amps
  - Bus volts - 115 volts
  - Charger Current - 0 amps

With the above conditions, the immediate mitigation strategy for DC bus 112 is to...

a. close Battery Charger 112 AC input breaker.

b. transfer DC bus 112 loads from "Normal" to "Reserve" power supplies.

c. shed non-essential DC bus 112 loads.

d. cross tie DC bus 112 to DC bus 212.

Answer:
d.

Explanation:
The question meets the K/A, requires examinee knowledge of operational implications of DC bus charger equipment and instrumentation during an event that would result in the loss of the DC bus if no action is taken.
When bus 142 is faulted, AC input power to battery charger 112 is de-energized. The DC bus loads will now be powered by the battery vs. the battery charger. It is reasonable to assume that a faulted 4kv bus is not going to be re-energized for several hours at a minimum. Therefore, in order to maintain the unit at power until such time as a controlled shutdown may be required, the crew must cross tie DC bus 112 to 212. 1BwOA ELEC-3 directs this action.
Choice A is incorrect, attempting to re-energize the battery charger will not work since it's ultimate power supply (bus 142 via bus 132X) is de-energized.
Choice B is incorrect, transferring DC loads to "reserve" feeds does not change the DC bus that they are powered from. Reserve feeds are simply alternate breakers and cabling that are still fed from the same DC bus as "normal" feeds.
Choice C is incorrect, load shedding is a contingency action that is taken when there is no alternative power supply to the DC bus.
Choice D is correct, 1BwOA ELEC-3 directs the crew to cross tie DC bus 112 with 212 which will place both bus loads on battery charger 212.

14 **Points: 1.00**

Given:

- Unit 1 was at 100% power, normal alignment.
- A loss of BOTH Unit 1 SX pumps has occurred.
- Currently the crew is taking actions to mitigate the event per the appropriate Abnormal Operating Procedure.
- The resulting pressure transient has caused tube leaks in the Unit 1 CC Hx.
- The Unit 1 CC surge tank level is slowly lowering.

With the above conditions, INITIAL CC surge tank makeup will occur...

a. automatically from the PW system.

b. ONLY when a local operator is dispatched to UNISOLATE the make up valves.

c. from the SX system IF normal SX pressure is restored prior to surge tank level reaching an auto make up setpoint.

d. automatically from the WM system.

Answer:
d.

Explanation:

Question meets KA - question requires examinee ability to monitor proper operation of CC surge tank level control during loss of SX event.

The CC surge tank make-up system was recently modified to address piping class concerns of the auto make up valves. During the mod process, the make up valves were manually isolated. However, the mod is now complete and the normal CC surge tank auto make up is restored. The initial make up will occur when surge tank level drops below 50% and the WM system make up valve will open. Continued level drop to 45% will open the PW system make up valve.

A is incorrect, auto make up from the PW system is the back up supply, not initial supply.

B is incorrect, this would have been correct during the plant mod of upgrading the piping class of the auto make up valves.

C is incorrect, restoring the SX system to normal pressure would slow the leak but not reverse it. SX system pressure is normally lower than CC system pressure.

D is correct, see explanation above.

15 **Points: 1.00**

Given:

- Braidwood Station has been notified that the State Estimator alarm predicts a potential degraded grid condition in the event of a Braidwood unit trip.

From which of the following locations can an NSO monitor specific switchyard bus voltages?

a. 0PM03J ONLY

b. 1PM01J ONLY

c. U-1 AEER ONLY

d. 0PM03J AND 1PM01J

Answer:
a.

Explanation:
The question meets the K/A, requires examinee ability to monitor grid frequency and voltage.
Grid voltage meters are located on 0PM03J panel in the MCR. 1PM01J contains the generator output voltage measured upstream of the MPTs. The AEER contains metering and transducers for grid potential circuits, but no voltage indication.
Choice A is correct, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

16 **Points: 1.00**

Given the following sequence of events:

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

The following indications are CURRENTLY noted on the MCB:

- RCS WR pressure is 1700 psig and stable.
- 1A RH discharge flow is 0 gpm.
- 1A SI pump discharge flow is 200 gpm.
- 1A CV pump flow is 300 gpm.

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

a. 1SI8801A, CHG PMPS TO COLD LEG INJ ISOL VLV

b. 1SI8821A, SI PP 1A TO COLD LEGS ISOL VLV

c. 1SI8809A, RH TO COLD LEG 1A & 1D ISOL VLV

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

Answer:
b.

Explanation:

The question meets the K/A, requires examinee knowledge of interrelations of a LOCA outside containment and components and functions of safety systems.
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 then step 2.d realigns systems as necessary.
Choice A is incorrect, CV pump flow is normal for current RCS pressure.
Choice B is correct, see explanation above.
Choice C is incorrect, RH pump flow is normal for current RCS pressure.
Choice D is incorrect, RH pump flow is normal for current RCS pressure.

17 **Points: 1.00**

Given:

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

- Bus 142 is faulted.
- ALL SG WR levels are 50%.
- RCS pressure is 2200 psig.
- Containment pressure is 1.4 psig.

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

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

a. RCS pressure rises to 2300 psig.

b. Containment pressure rises to 5.2 psig.

c. 1A CV pumps trips.

d. ALL SG WR levels drop to 35%.

Answer:
c.

Explanation:

The question meets the K/A, requires examinee knowledge of limiting conditions for operation (loss of a ESF bus and a ECCS pump) and how that could relate to a loss of heat sink event.
1BwFR-H.1 OAS page list bleed and feed criteria after performance of step 3 as any of the following.
WR SG level <27% (43% adverse) in any 3 SGs.
RCS pressure >2335 due to loss of heat sink.
No CV pumps available.
Choice A is incorrect, pressure still below 2335#.
Choice B is incorrect, containment would be adverse, but current SG levels at 50% would not meet bleed and feed criteria.
Choice C is correct, no CV pump available with 1A CV pump tripped combined with loss of bus 142.
Choice D is incorrect, SG levels still above feed and bleed criteria (27%).

18 **Points: 1.00**

Given:

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

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

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

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

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

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

Answer:
a.

Explanation:
The question meets the K/A, requires examinee knowledge of the operational implications during a uncontrolled depressurization of all SGs as it applies to emergency systems (AF system).
The 2AF005 valves can be controlled with IA controllers from the RSP that bypass the electrical signal from the MCR controllers. Therefore the valve can be controlled from the RSP. Also the 2AF005 valves have local handwheels that allow local throttling or closing. With the given information 2BwCA-2.1 would require 45 gpm flow be established to each SG. With 2A AF pp not available and bus 142 faulted (B train 2AF013 valves without power) the 2AF005H valve would need to be throttled either at the RSP or locally.
Choice A is correct, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, 2AF013D is A train and 2A AF pump is tripped.
Choice D is incorrect, 2AF013H does not have power with bus 242 de-energized.

19 **Points: 1.00**

Given:

- Unit 1 reactor tripped from full power.
- 2 control rods did NOT fully insert.
- A U-1 SAT fault created a loss of offsite power concurrently with the reactor trip.
- All other equipment operated as designed.
- The ONLY operator actions taken so far were manual reactor trips from 1PM05J and 1PM06J.

With the above conditions, which of the following sets of MCR actions will correctly initiate emergency boration of the RCS in accordance with 1BwOA PRI-2, EMERGENCY BORATION.

a. 1. OPEN 1CV112D, RWST TO CHG PPS SUCT VLV.
   2. CLOSE 1CV112B, VCT OUTLET ISOL VLV.
   3. MANUALLY raise controller 1FK-121, CENT CHG PMPS FLOW CONT VLV, demand.

b. 1. OPEN 1CV8104, EMER BORATION VLV.
   2. START 1AB03P, BORIC ACID XFER PP.

c. 1. CLOSE 1CV112B, VCT OUTLET ISOL VLV.
   2. OPEN 1SI8801A, CHG PMPS TO COLD LEG INJ ISOL VLV.

d. 1. OPEN 1CV110A, BORIC ACID TO BLNDR VLV.
   2. OPEN 1CV110B, BORIC ACID BLNDR TO CHG PMPS VLV.
   3. START 1AB03P, BORIC ACID XFER PP.

Answer:
a.

Explanation:
The question meets the K/A, requires examinee ability to determine if control room switched correctly reflect desired plant line-up.
A failure of 2 RCCAs to fully insert requires emergency boration of the RCS. With a loss of off site power the boric acid transfer pump will not have power because it is powered by non-ESF bus. Therefore the only option to borate is from the RWST by aligning CV pump suction to the RWST. The discharge path of the CV pumps will be through the normal charging header via 1CV121 by raising flow on 1CV121 to max on scale. An optional discharge path is through the high head SI header via 1SI8801A/B.
Choice A is correct, see explanation above.
Choice B is incorrect, no power available to boric acid pump.
Choice C is incorrect, this method does not open a suction path to the CV pump.
Choice D is incorrect, no power available to boric acid pump.

20 **Points: 1.00**

Which of the following sets of nuclear instrumentation readings, if noted during a reactor startup, would indicate one or more inoperabilities of the given instruments?

	SRNI N-31	SRNI N-32	IRNI N-35	IRNI N-36
a.	$1x10^{0}$ cps	$1x10^{0}$ cps	$6x10^{-10}$ amps	$4x10^{-9}$ amps
b.	$5x10^{2}$ cps	$2x10^{3}$ cps	$1x10^{-11}$ amps	$1x10^{-11}$ amps
c.	$9x10^{2}$ cps	$1x10^{4}$ cps	$2x10^{-11}$ amps	$1x10^{-10}$ amps
d.	$9x10^{3}$ cps	$2x10^{4}$ cps	$9x10^{-11}$ amps	$1x10^{-10}$ amps

Answer:
c.

Explanation:

Question meets KA. Question requires examinee knowledge of the maximum allowable channel disagreement.

The surveillance requirement for source range and intermediate range NIs is a daily channel check. Per 1BwOSR 0.1-1,2,3, attachment A, the maximum channel deviation between redundant instruments is 1 decade (at one half decade difference, an evaluation is made based upon IM readings).

A is incorrect, source ranges are pegged low, but that would be normal indication when intermediate range is above P-6 and source ranges are blocked.

B is incorrect, intermediate ranges are pegged low, but that would be normal for power level of source ranges (IRNIs are not on scale yet).

C is correct, difference between N-31 and N-32 is greater than one decade.

D is incorrect, both source and intermediate ranges are reading within one decade of redundant instrument.

21 **Points: 1.00**

Given:

- A fire is occurring in the Upper Cable Spreading Room directly above the Unit 1 MCR.
- All detectors in the area are in alarm on 1PM09J, Fire Detection panel.

Main Control Room operator response regarding fire suppression to this event is to...

a. verify automatic Halon suppression actuation.

b. verify automatic CO_2 suppression actuation.

c. verify proper fire pump response to automatic sprinkler system actuation.

d. dispatch fire brigade members to manually actuate foam deluge system.

Answer:
a.

Explanation:
The question meets the K/A, requires examinee knowledge of operational implications of plant fire by fire type (electrical/cable tray fire)
Halon will auto actuate on 2/2 coincidence of POC and thermal detectors in the upper cable spreading room. CO2 backup system has been abandoned in this area. There is no sprinkler or foam system installed in this area since they would not be a desirable quenching agent to use on a electrical cable fire.
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.

22 **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.

Explanation:
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. 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 (logic) 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 student to confuse power supplies and logic cabinets with control banks and groups.
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.

23 **Points: 1.00**

Given:

- Unit 1 had a reactor trip and safety injection due to an inadvertent SI relay actuation.
- ALL other plant equipment operated as expected.
- Currently, RCS temperature is being maintained stable by the steam dumps in Tave Mode.
- Aux. Feedwater flow has been throttled to maintain stable SG levels.
- The crew has just entered 1BwEP ES-1.1, SI TERMINATION.

Compared with current steam dump demand, after 1BwEP ES-1.1 is complete, steady state steam dump demand will be... (assume RCS decay heat load is constant)

a. lower, due to transferring steam dumps to steam pressure mode.

b. the same, since decay heat load did NOT change.

c. higher, due to the CVCS system realignment to normal charging and letdown.

d. higher, due to securing the RH pumps.

Answer:
c.

Explanation:

The question meets the K/A, requires examinee knowledge of interrelations between SI termination and facility heat removal systems.
Following SI termination, the reduction in ECCS flow will cause less decay heat removal from ECCS that must be compensated for with other heat removal systems. If steam dumps are maintaining RCS temp at no load temperature in auto, then steam dump demand would rise when the CV pumps are realigned for normal charging and letdown to account for the decay heat load that is no longer being removed by ECCS flow.
Choice A is incorrect, transferring to stm pressure mode is performed during 1BwEP ES-1.1. However, auto setpoint of steam pressure mode is 1092 psig, which is the saturation pressure for 557°F, therefore demand should return to the same value given that steam dumps were controlling at 557°F in tave mode prior to SI termination.
Choice B is incorrect, although decay heat not changing is assumed in stem, less decay heat would be removed by ECCS flow so steam dumps would open further to compensate.
Choice C is correct, see explanation above.
Choice D is incorrect, the RH pumps would have be running on recirc only during a inadvertent SI because RCS pressure would have remained above the RH pp dead head value.Therefore securing RH pps should not affect decay heat load.

24 **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 RCS pressure drops during the cooldown, which of the following is the EARLIEST indicated RCS 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.

Explanation:
The question meets the K/A, requires examinee ability to operate/monitor instrumentation and interlocks during an post LOCA Cooldown and Depressurization.
1BwEP ES-1.2, step 8.c has the crew check RCS pressure <1930 psig prior to blocking Low Steamline Pressure SI to prevent the MSIV from closing. This cannot be performed until below the P-11 setpoint. Typically when the operating crew reaches this point in the simulator, the RCS pressure is still above P-11. The RNO column has the crew continue with the procedure and come back to block the SI signal when RCS pressure is below P-11.
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 to 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.

25 **Points: 1.00**

Given the following plant conditions on Unit 1:

- A pressurizer pressure control malfunction has resulted in a saturated core cooling condition.
- 1BwFR-C.3, RESPONSE TO SATURATED CORE COOLING, is in progress.

During performance of 1BwFR-C.3, step 3, CHECK RCS VENT PATHS, the correct RO action concerning the PZR PORVs is to have ...

a. ONE PORV OPEN to promote RCS natural circulation flow.

b. ONE PORV OPEN to promote SI accumulator injection.

c. BOTH PORVs CLOSED to reduce RCS inventory loss.

d. BOTH PORVs OPEN to enhance ECCS flow.

Answer:
c.

Explanation:
The question meets the K/A, requires examinee knowledge of reasons for RO function within the control room to maintain procedure adherence.
The mitigation strategy for a saturated core cooling condition is to raise ECCS flow (by aligning injection flow path) and stopping any RCS inventory loss from the PZR PORVs or Rx head vents. The question is intended to test the basic mitigation strategy of how PZR PORVs are operated in a saturated core cooling event vs. another type of event.
Choice A is incorrect, SG PORVs are used to dump steam during a loss of natural circ event, however opening PZR PORVs would not enhance, but would likely degrade natural circulation.
Choice B is incorrect, depressurizing the RCS to get SI accumulators to inject is a strategy for an inadequate core cooling event, but is done by depressurizing the SGs via steam dumps or SG PORVs.
Choice C is correct, see explanation above.
Choice D is incorrect, opening both PZR PORVs to enhance SI flow (feed and bleed) is a mitigation strategy for a loss of heat sink event.

26 **Points: 1.00**

Given:

- Unit 1 is experiencing an anticipated reactor vessel pressurized thermal shock event.
- Performance of 1BwFR-P.1, RESPONSE TO IMMINENT PRESSURIZED THERMAL SHOCK is in progress.
- ECCS flow has been terminated and the crew is checking if ECCS re-initiation is required.
- RCS subcooling is UNACCEPTABLE.
- RCS inventory is slowly lowering.

As inventory drops, which of the following is the MAXIMUM RCS inventory level that would require the crew to manually start and re-align ECCS pumps?

a. Pressurizer level of 10%

b. RVLIS HEAD level of 31%

c. RVLIS PLENUM level of 37%

d. RVLIS PLENUM level of 0%

Answer:
d.

Explanation:
The question meets the K/A, requires examinee knowledge of operational implications of remedial actions associated with pressurized thermal shock.
FR-P.1 has less restrictive RCS inventory ECCS reinitiation criteria than most emergency procedures because the ECCS flow may be contributing to the excessive RCS cooldown and pressurization.
The SI reinitiation criteria in 1BwFR P.1 CAS page is adequate subcooling per iconics or attachment A and RVLIS plenum region $\geq$15%. 1BwFR-P.1 WOG background document states "due to the less restrictive reinitiation criteria the operator should be especially alert for changes in subcooling and RCS inventory".
Choice A is incorrect, but plausible because 1BwEP ES-1.1 OAS page has ECCS reinitiation criteria of pressurizer level <14%. Therefore 10% PZR level would meet this criteria.
Choice B is incorrect, but plausible because 1BwEP ES-0.1 OAS page has ECCS initiation criteria of pressurizer level <4%. Therefore 37% RVLIS head would meet this criteria.
Choice C is incorrect, but plausible because 1BwCA 3.2 OAS page has ECCS reinitiation criteria of RVLIS plenum <55%. Therefore 31% RVLIS plenum level would meet this criteria.
Choice D is correct, RVLIS PLENUM indication of 0% is the next lower indication below 15% plenum region (see explanation above).

27 **Points: 1.00**

Given:

- An event has occurred on Unit 1 causing elevated containment radiation.
- The crew has entered 1BwFR Z.3, RESPONSE TO HIGH CONTAINMENT RADIATION LEVEL.

To reduce containment radiation level in accordance with 1BwFR-Z.3, the operators may start the...

a. Containment Spray System.

b. Containment Charcoal Filter System.

c. Containment Mini-Flow Purge System.

d. Reactor Containment Fan Cooler System.

Answer:
b.

Explanation:
The question meets the K/A, requires examinee ability to operate or monitor the operating behavior characteristics of the plant as they apply to high containment radiation.
1BwFR-Z.3 mitigation strategy is to ensure containment ventilation is isolated, then evaluate usage of the containment charcoal filter units and post LOCA purge system by the TSC.
Choice A is incorrect, CS system is not a choice for lowering rad levels in 1BwFR-Z.3, although the WOG background document refers to CS system as considered to reduce containment rads, but not chosen since its primary purpose is to remove heat during high containment pressure events.
Choice B is correct, see explanation above
Choice C is incorrect, containment mini-flow purge system is not a choice for lowering rad levels in 1BwFR-Z.3, however post LOCA purge is mentioned for evaluation by the TSC, yet no specific procedure steps exist in FR-Z.3 to put the post LOCA purge system in operation.
Choice D is incorrect, RCFC system is not a choice for lowering rad levels in 1BwFR-Z.3, but is used for containment cooling in accident events.

28 **Points: 1.00**

If an RCP Oil Lift Pump Pressure Switch fails at 500 psig, the RCP __(1)__ start and _____(2)_____.

	(1)	(2)
a.	WILL	WILL NOT trip once started
b.	WILL NOT	WILL trip if previously running
c.	WILL	WILL TRIP after a time delay
d.	WILL NOT	WILL NOT trip if previously running

Answer:
d.

Explanation:

Question meets KA - question requires examinee knowledge of interlocks which provide for adequate lubrication of the RCP.

An RCP requires 600 psig lift oil pressure as a starting interlock. However, the interlock is only in the starting circuit, not in the trip coil circuit. Therefore a failure of the pressure switch to 500 psig will prevent the pump from starting, but not trip a previously running pump.

A is incorrect, pump will not start.
B is incorrect, pump will not trip.
C is incorrect, pump will not start.
D is correct, see explanation above.

29 **Points: 1.00**

Given:

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

The following then occurs:

- 1CC9416 CC FROM RC PUMPS ISOL VLV, inadvertently CLOSES and will NOT re-open.

With the above conditions, RCP...

a. motor bearing temperatures will rise and RCP temperatures should be monitored for motor bearing trip criteria of 195°F maximum.

b. motor bearing temperatures will rise and RCP temperatures should be monitored for motor bearing trip criteria of 225°F maximum.

c. seal outlet temperatures may rise and RCP temperatures should be monitored for seal outlet trip criteria of 235°F maximum.

d. seal outlet temperatures may rise and CC Surge Tank level should be monitored for adequate make-up.

Answer:
a.

Explanation:

The question meets the K/A, requires examinee ability to predict the impact of problem associated with bearing RCP bearing temperatures and use procedures to mitigate the affect.

The Component Cooling Water supply to the RCP motor bearings and thermal barriers has a common supply line and separate return lines. 1CC9416 is the containment isolation valve on the return line from the motor bearings only. Therefore, closing 1CC9416 will isolate cooling water flow from the RCP motor bearings only.

RCP motor bearing temperature trip criteria of 195°F, appears in multiple procedures including 1BwOA PRI-6 step 4, for monitoring RCP temperatures, in BwOP RC-1 (RCP startup procedure) limitations and actions and a caution, and 1BwGP 100-1 (Plant Heatup procedure) cautions. Additionally, Bwd training objective S.RC2-09-E requires the candidate to know the conditions and setpoints requiring RCP trip.

Choice A is correct, see explanation above.

Choice B is incorrect, 225°F is the pump lower radial bearing temp trip setpoint.

Choice C is incorrect, but would be plausible if the examinee assumes that cooling flow is isolated from the thermal barriers. 235°F is the seal outlet temp trip setpoint.

Choice D is incorrect, but would be plausible if the examinee assumes that some cooling flow may still exist via the relief valve lifting on the CC lines inside containment. A similar situation happens when the RCP seal return flow containment isolation valve is closed. Seal return flow will not cease, rather the relief valve on the seal return line lifts and re-directs that flow to the containment sump. If the examinee assumes a similar result on the CC outlet line, the CC surge tank level monitoring would be prudent due to the assumed loss of CC water to the containment sump. However, this will not happen in the CC system because the relief valve setpoint of 150 psig is higher than the CC pump supply pressure.

30 **Points: 1.00**

Given:

- Unit 1 is in Mode 5.
- Pressurizer is in a solid condition.
- 1CV128, RH TO CV LTDWN FLOW CONT VLV, is open.
- 1CV131, LTDWN LINE PRESSURE CONT VLV, is in automatic, controlling RCS pressure at 350 psig.
- 1A RH train is in standby.
- 1B RH train is in shutdown cooling.

If 1PK-131, LTDWN LINE PRESSURE CONT VLV 1CV131 controller "set point" dial is slowly turned ONE revolution CLOCKWISE, 1CV131 valve will throttle ___(1)___ and RCS pressure will ___(2)___.

a. (1) OPEN
(2) DROP

b. (1) CLOSED
(2) RISE

c. (1) OPEN
(2) RISE

d. (1) CLOSED
(2) DROP

Answer:
b.

Explanation:
Question meets K/A - requires examinee ability to predict changes in RCS pressure associated with operating CVCS controls.
The 1CV131 controller has a setpoint dial of that will rotate 10 times from 0 - 10 position with a range of 0 - 600 psig. With the original setting of 350 psig, the controller demand will be set at approx. 5.8. Turning the controller one revolution clockwise will demand a rise in letdown backpressure causing 1CV131 valve position to throttle close and raise RCS pressure approx. 60 psig to the new controller 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.

31 **Points: 1.00**

Given:

- Unit 2 is in Mode 5.
- All systems are normally aligned.
- RCS pressure is 300 psig.
- 2A RH pump is running in the shutdown cooling mode per BwOP RH-6.

The following occurs:

- A 300 gpm tube leak develops in the 2A RH Heat Exchanger.

Based on the above conditions and assuming NO operator actions, what is the expected response of the following 2PM06J instruments?

2FI-0689, CC TO RH HX 2A FLOW, INDICATED inlet flow will _______.
2FK-618, HX 2A BYP FLOW CONT VLV 2RH-0618, DEMAND position will ________.

	2FI-0689	2FK-618
a.	drop	rise
b.	drop	drop
c.	rise	rise
d.	rise	drop

Answer:
a.

Explanation:

Question meets KA - question requires examinee knowledge of the effect an RH HX malfunction will have on the RH system.
A RH HX tube leak with the RCS at 300 psig would cause flow into the CC system raising pressure on the shell side of the RH heat exchanger. This would cause less CC flow into the heat exchanger. Because the CC to RH HX flow orifices are on the inlet to the heat exchanger, indicated flow would drop. Also, the loss of RH flow through the leak would cause total RH return flow to drop. 2RH0618 controller will sense the drop in flow and raise demand automatically to try to maintain 3300 gpm total RH flow returning to the RCS. The question is a 1 of 2 x 2 format, but both parts require detailed system knowledge and are high cognitive level.
A is correct, see explanation above.
B is incorrect, see explanation above.
C is incorrect, see explanation above.
D is incorrect, see explanation above.

32 **Points: 1.00**

Given:

- Unit 1 is in Mode 5 following refueling.
- All system are normally aligned.
- 1B RH pump is running in shutdown cooling mode.
- An RCS heatup is in progress in preparation for a mode change scheduled next shift.

The following occurs:

- A fire forces a Main Control Room personnel evacuation to the Remote Shutdown Panel (RSP).
- 1BwOA PRI-5, CONTROL ROOM INACCESSIBILITY, is implemented.
- After activating the RSP, the crew determines that 1RH0607, HX 1B FLOW CONT VLV, needs to be throttled to stop the RCS heatup and prevent an inadvertent mode change.

With the above conditions, the crew will throttle 1RH0607 by...

a. dispatching an EO to the 1B RH HX Room to isolate control air and gag the valve open.

b. dispatching an EO to the 364' Aux Bldg to bypass the valve positioner with pneumatic jumpers.

c. dispatching an IM to the Aux Elec. Equip. Room to bypass the valve controller with electric jumpers.

d. manually operating the valve with a controller on the Remote Shutdown Panel.

Answer:
b.

Explanation:
Question meets KA - question requires examinee knowledge of local auxiliary operator actions of the RH system during an emergency .
Under a MCR evacuation 1BwOA PRI-5 directs the crew establish a cool down of the RCS by throttling the RH HX FCVs with a temporary regulator and pneumatic jumper.
A is incorrect, the RH FCVs do not have installed manual gags similar to the the HD FCVs or the AF FCVs.
B is correct, see explanation above.
C is incorrect, IMs installing jumpers to test connections is a common method for calibrating instrumentation, but is not used in 1BwOA PRI-5.
D is incorrect, this distractor is plausible because several valves (1AF005s, 1CV121) have controllers in the RSP, but RH FCVs are not included on the RSP.

33 **Points: 1.00**

Given the following sequence of events:

- Unit 1 was at 100% power, normal alignment.
- An RCS LOCA and auto SI occur.
- During performance of 1BwEP ES-1.3, TRANSFER TO COLD LEG RECIRCULATION, the 1SI8804B, RH HX 1B TO SI PMPS ISOL VLV, would NOT open.

With the above condition, the cold leg recirc alignment status of the U1 SI pumps will be...

a. NEITHER pump will have a recirc suction source and BOTH MUST be shutdown when RWST level reaches 9%.

b. ONLY the 1A pump will have a recirc suction source and the 1B pump MUST be shutdown when RWST level reaches 9%.

c. ONLY the 1B pump will have a recirc suction source and the 1A pump MUST be shutdown when RWST level reaches 9%.

d. BOTH pumps will have a recirc suction source and can remain running after isolation from the RWST.

Answer:
d.

Explanation:
Question meets K/A - requires examinee knowledge of how a valve malfunction will effect ECCS system. The ECCS system design contains a suction crosstie header between the CV and SI pumps, such that either RH pump can supply suction to all the CV and SI pumps. Therefore the failure of the 1SI8804B to open would not prohibit running either SI pump in the recirc mode. Although the 1SI8804B supply header connects with the 1B SI pump suction and the suction crosstie header connects with the 1A SI pump suction, because the SI pump suctions are also crosstied via their suction valves 1SI8923A and B, both pumps can remain in service on recirc with a failure of either the 1SI8804B or 1CV8804A.
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

34 **Points: 1.00**

Given the following sequence of events:

- Unit 1 was at 100% power, normally aligned.
- An inadvertent relay actuation causes an SI and Phase A Containment Isolation.
- All other plant equipment operates as designed.

With the above condition, if ECCS flow is NOT terminated, 10 minutes after the relay actuation, maximum RCS pressure will be controlled by...

a. Pressurizer PORV 1RY455A opening.

b. Pressurizer PORV 1RY456 opening.

c. Pressurizer Safety Valves opening.

d. ECCS pump design shutoff head pressure.

Answer:
b.

Explanation:
Question meets K/A - requires examinee ability to monitor overpressure protection system as it relates to ECCS.
With an inadvertent SI from full power, RCS pressure initially drops due to the reactor trip and subsequent cooldown to hot zero power Tave. However, within approx. 5 minutes the rising RCS inventory due to ECCS injection flow will overpressurize the RCS. Although the phase A containment isolation will cause a loss of instrument air in containment, the pressurizer PORVs are equipped with instrument air accumulators sized to allow > 50 PORV cycles with air isolated to containment. The 50 cycles will allow the 1RY456 PORV to control RCS pressure well beyond the 10 minutes bounded in the question stem. The 1RY456 will cycle before the 1RY455A PORV because its lift setpoint bistable is set 10 psi lower (2335 psig vs 2345 psig).
Choice A is incorrect, because 1RY455A lift setpoint is 10 psi higher than 1RY456.
Choice B is correct, see explanation above.
Choice C is incorrect, because the PORVs have instrument air accumulators allowing them to open after instrument air is isolated to containment and the PORV lift setpoints are below the safety lift setpoints of 2460 psig.
Choice D is incorrect, because the ECCS (CV) pump design shutoff head is 2600 psig which is above the PORV and safety setpoints

35 **Points: 1.00**

Given:

- Unit 1 is at 100% power.
- All systems are normally aligned.

The following occurs:

- A Pressurizer PORV fails open.
- The associated PZR PORV Block valve will NOT close.
- Operators perform a manual reactor trip and safety injection.
- The NSO has just reported the PRT rupture disk has blown.

In addition to lowering PRT pressure, which of the following indications is consistent with the PRT rupture disk opening? (assume NO other failures have occurred)

a. Containment pressure is RISING at 5 psig per minute.

b. PRT level is LOWERING quickly.

c. RCFC amps are LOWERING.

d. Containment RF sump leak detection flow rates are RISING.

Answer:
d.

Explanation:

Question meets KA - question requires examinee knowledge of effect of PRT rupture on containment.
A is incorrect, containment pressure rise is not solely indicative of PRT rupture and rate would not be 5 psig per minute from a PRT rupture.
B is incorrect, the PRT rupture disks are on the top of the PRT. When it ruptures there may be a minor level drop due to more flashing because of the rapid pressure drop but it will be very slight.
C is incorrect, cnmt atmosphere would be getting more dense, causing RCFC amps to rise.
D is correct, sump flow rate would rise as a result of the steam and condensation input to cnmt.

36 **Points: 1.00**

Given:

- Unit 1 is at full power, normal alignment.
- The 1A CC pump is running.
- The 1B CC pump is in standby with its C/S is in NAT.
- The U-0 CC pump is in standby, aligned to bus 142 with its C/S in NAT.

The following occurs:

- Breaker 1412 is inadvertently opened from the MCR.
- All equipment operates as designed.

With the above conditions and NO further operator action, ONE minute later the status of the CC pumps will be...

	U-0 pump	1A pump	1B pump
a.	running	running	running
b.	stopped	running	stopped
c.	stopped	running	running
d.	stopped	stopped	running

Answer:
c.

Explanation:

Question meets K/A, requires examinee knowledge of CC system design interlocks which provide for automatic start of the standby pump.

Upon the inadvertent trip of 1412 breaker, bus 141 will de-energize and the undervoltage relays will strip the bus loads and start the 1A DG. Four seconds after a low CC pressure condition is reached, the 1B CC pump will start on a low system pressure signal. The U-0 pump will be blocked from starting because the 1B pump C/S is not in PTL. The 1A DG will start and re-energize bus 141. Twenty seconds after the bus is re-energized the 1A CC pump will restart on the ESF sequencer.

Choice A is incorrect, this would be correct if the U-0 pump also started on low system pressure, however that feature is blocked unless the 1B pump is in PTL.

Choice B is incorrect, this would be correct if the low pressure start feature was blocked by the DG carrying the bus in emergency mode. This feature does exist however it is train specific and the 1A DG will not block the pumps being supplied by bus 142.

Choice C is correct, see explanation above.

Choice D is incorrect, this would be correct if assumed that the 1B CC pump starts on low pressure and then normal system pressure blocks the 1A pump from restarting.

37 **Points: 1.00**

Given:
- Unit 1 is at 75% power, normally aligned.
- All systems are at stable, steady state condition.

Then the following event occurs:
- A charging flow control malfunction causes pressurizer level to rise.
- The crew takes manual control of charging flow and stabilizes pressurizer level at 60%.
- ALL other systems remain in automatic control and the crew maintains pressurizer level at 60% until stable, steady state conditions exist again.

With the above conditions, compare STEADY STATE demand on the on 1PK-455A, MASTER PZR PRESS CONTROLLER, BEFORE and AFTER the event.

AFTER the event, 1PK-455A DEMAND will be...

a. HIGHER because of more pressurizer heater input.

b. LOWER because a smaller pressurizer bubble exists.

c. LOWER because spray flow is more effective.

d. the SAME because saturation pressure is the same.

Answer:
a.

Explanation:
The question meets the K/A, requires examinee knowledge of cause/effect relationship between the pressurizer PCS and LCS.
At initial conditions of 75% power, program pressurizer level is 51.25%. A transient that raised level to 60% at the same power level will cause a >5% deviation in pressurizer level. This will automatically turn on all the back-up heaters and greatly raise the heat input into the pressurizer. The backup heaters have a capacity of 1400 Kw vs. only 400 Kw for the variable heaters. This rise in heat input will cause the master pressurizer pressure controller demand to rise, lowering variable heater demand and raising spray flow demand, to maintain the pressurizer at 2235 psig steady state condition.
Choice A is correct, see explanation above.
Choice B is incorrect, during the transient, the rising liquid level will squeeze the pressurizer bubble and raise system pressure, however this will act to raise demand on the master pressure controller.
Choice C is incorrect, Although spray may be slightly more effective with a smaller bubble, initial steady state conditions have zero spray flow (except for a small amount of bypass flow) also turning on the back up heaters significantly raises heat input and therefore demand for spray flow.
Choice D is incorrect, although saturation pressure is the same, the rise in heater input causes higher demand on the master pressurizer pressure controller to maintain the same saturation conditions.

38 **Points: 1.00**

Given:

- Unit 1 was at 100% power, all systems normally aligned.
- 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 the position indication lights.

b. and from TSLB-4 (but NOT from the position indication lights).

c. and from the position indication lights (but NOT from TSLB-4).

d. but NOT from either the position indication lights OR TSLB-4 .

Answer:
b.

Explanation:

Question meets K/A, requires examinee ability to monitor automatic operation of the reactor protection system circuit breakers.

Power to the position indication lights 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 correct, see explanation above.
Choice C is incorrect, see explanation above.
Choice D is incorrect, see explanation above.

39 **Points: 1.00**

Which of the following reactor protection system trips serves as a BACK-UP to the Power Range Neutron Flux - High trip and is designed to ensure that the allowable heat generation rate (kw/ft) of the fuel is NOT exceeded?

a. Pzr low pressure

b. OPDT

c. RCS low flow

d. Pzr high pressure

Answer:
b.

Explanation:
Question meets K/A, requires examinee knowledge of operational implication of concept of power density as it relates to the reactor protection system.
Choice A is incorrect, Pzr low pressure trip setpoint is designed to prevent exceeding DNBR limits.
Choice B is correct, OPDT trip setpoint is designed to prevent exceeding peak fuel centerline temperature at high power.
Choice C is incorrect, RCS low flow trip setpoint is designed to prevent exceeding DNBR limits.
Choice D is incorrect, Pzr high pressure trip setpoint is designed to prevent RCS Integrity limits.

40 **Points: 1.00**

Given:

- Unit 1 was at 100% power, normal alignment.
- Subsequently, containment pressure transmitter 1PT-0934 failed.
- ALL required actions of Tech Specs have been completed.

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

a. 1 of 2.

b. 2 of 2.

c. 1 of 3.

d. 2 of 3.

Answer:
d.

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

41 **Points: 1.00**

Given:

- Unit 1 was at 100% power, normally aligned.
- A DC circuit breaker trip causes annunciator 1-21-B10, BUS 131X CONT PWR FAILURE, to alarm.
- NO other annunciators are in alarm.

With the above condition, the U-1 train "A" RCFC fans _____(1)_____ and, if a U-1 safety injection occurs these fans WILL _________(2)___________ in LOW speed.

	(1)	(2)
a.	WILL trip from HIGH speed	NOT start in auto or manual
b.	REMAIN running in HIGH speed	NOT start in auto or manual
c.	WILL trip from HIGH speed	auto start
d.	REMAIN running in HIGH speed	auto start

Answer:
b.

Explanation:

The question meets the K/A, requires examinee knowledge of power supplies to containment cooling fans. Alarm 1-21-B10 indicates a loss of DC control power to bus 131X which would disable the auto functions of the breakers and prevent any remote (manual) functions. Therefore the RCFCs would remain running in high speed and could not be started in auto or manual in low speed. The only way to trip or start an affected fan would be locally at the circuit breaker.

Choice A is incorrect, Hi speed breaker does not trip.
Choice B is correct, see explanation above.
Choice C is incorrect, Hi speed breaker does not trip, Low speed breaker will not auto start.
Choice D is incorrect. Low speed breaker will not auto start.

42 **Points: 1.00**

Given:

- Unit 1 is 100% power.
- The 1A CS pump is currently running on recirc for an ASME surveillance per BwOP CS-5, CONTAINMENT SPRAY SYSTEM RECIRCULATION TO THE RWST.

Which of the following would cause the fastest heat up of the 1A CS pump? (analyze each separately)

a. 1SI001A, CS PP 1A DSCH TEST LINE ISOL TO RWST VLV, mechanically fails CLOSED.

b. 1CS010A, EDUC 1A INLET FLOW CONT VLV, electrically fails CLOSED.

c. U-1 experiences a LOSS of all Component Cooling.

d. U-1 experiences a LOSS of all Essential Service Water.

Answer:
a.

Explanation:

Question meets the K/A, requires examinee ability to predict changes in parameters which would affect CS pump cooling.

When a CS pump is running on recirc the pump cooling is provided by the forward flow through the pump with the RWST being the heat sink. If 1SI001A closed during a recirc run, the CS pump flow would be restricted to the eductor line flow path which is in effect a pump recirc line but it has no heat sink. This would cause the fastest heat up of the pump.

Choice A is correct, see explanation above.

Choice B is incorrect, if 1CS010A closed, eductor flow would stop, however the primary flowpath of recirc back to the RWST would still cool the pump.

Choice C is incorrect, there is no interface between the CC system and CS system for pump cooling. Although, the CS pumps are very similar to the RH pumps (which have a seal cooler that is cooled by CC), the CS pumps do not have that feature since they are not designed for plant cool down.

Choice D is incorrect, the SX system provides cooling to the CS pump cubicle cooler which is designed to maintain the CS room environment during accident conditions, however this would primarily affect the CS pump motor cooling vs. pump cooling.

43 **Points: 1.00**

Given:

- Unit 1 is at 100% power, normal alignment.
- A grid disturbance causes the main turbine to reach the overspeed protection circuit setpoint.

When the OPC 20-1 and 20-2 solenoid valves open, the Main Turbine ________________________ valves will reposition closed to limit turbine speed.

a. governor and intercept

b. throttle and intercept

c. governor and reheat stop

d. throttle and reheat stop

Answer:
a.

Explanation:
The question meets the K/A, requires examinee knowledge of the cause/effect relationship between the main steam system and the turbine/generator system.
When the OPC solenoids open on a turbine overspeed signal, the DEHC Aux Governor trip header will de-pressurize and throttle closed the governor and intercept valves until turbine speed drops below the setpoint. Then the OPC solenoids will de-energize and close. This will re-pressuize the aux governor header and reopen the GVs and IVs.
Choice A is correct, see explanation above.
Choice B is incorrect, throttle valves do not close.
Choice C is incorrect, reheat stop valves do not close.
Choice D is incorrect, throttle and reheat stop valves do not close.

44 **Points: 1.00**

Given:

- Unit 1 is at 85% power.
- The crew is performing 1BwOSR TRM 3.3.g.4, TURBINE OVERSPEED PROTECTION SYSTEMS VALVE STEM FREEDOM CHECKS (TV-GV CYCLING).

Which of the following describes system performance during the Governor Valve #1 (GV1) stroke test?

As GV1 strokes closed...

a. GV2, GV3 and GV4 stroke open (all) together to maintain turbine load constant.

b. GV2 and GV3 stroke full open together, then GV4 strokes open (independently) to maintain turbine load constant.

c. GV2, GV3 and GV4 sequentially (all independently) stroke open to maintain turbine load constant.

d. GV2, GV3 and GV4 remain stationary and turbine load drops with steam demand.

Answer:
a.

Explanation:
The question meets the K/A, requires examinee knowledge of a main steam system surveillance procedure.
1BwOSR TRM 3.3.g.4 can be performed with the turbine on line or off line which changes how the test is conducted. Additionally, the GVs are tested differently at power than the TVs to minimize the amount of turbine load change that occurs. The operator should understand the system operating mode during each different test to verify proper system operation. When testing GVs at power, the surv. directs the system to be placed in single valve mode prior to the test. Therefore, when a GV is stroked closed, the remaining 3 valves will stroke open together to maintain turbine load constant.
Choice A is correct, see explanation above.
Choice B is incorrect, this describes how GVs would react if in sequential valve mode.
Choice C is incorrect, this is a common misconception of how GVs react if in sequential valve mode.
Choice D is incorrect, this describes how a TV test takes place.

45 **Points: 1.00**

Given:

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

The following event occurs:

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

With the above conditions, the NSO will...

a. TRIP a running FW pump.

b. START the standby HD pump.

c. OPEN 1CB025, LP HTR BYP VLV.

d. START the standby CD/CB pump.

Answer:
d.

Explanation:

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

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

Choice A is incorrect, this is plausible because a similar alarm (1-16-D1, Low FW PP Suct Hdr Press) has an auto action that trips the 1A FW pump.

Choice B is incorrect, this is plausible because a probable cause of the low NPSH condition in BwAR 1-16-E1 is insufficient HD pps running. However, there is no auto start feature for the HD pps.

Choice C is incorrect, this is plausible because bypassing the LP heater string is one possible method of getting more flow to the FW pp suction. It is also an auto action of a HI-2 level condition in a 11 heater.

Choice D is correct, the first auto action listed in BwAR 1-16-E1 is start of standby CD/CB pump.

46 **Points: 1.00**

Compare and contrast the conditions below for Unit 1:

- A rapid addition of feedwater is made to the 1B S/G at 6% power.
- A rapid addition of feedwater is made to the 1B S/G at 80% power.
- (assume the feedwater additions are identical)

The 1B S/G's INITIAL indicated level response to the rapid addition of feedwater is to ___(1)___ with a greater magnitude of change at ___(2)___ power.

	(1)	(2)
a.	shrink	low
b.	shrink	high
c.	swell	low
d.	swell	high

Answer:
a.

Explanation:

The question meets the K/A, requires examinee ability to predict impact of overfeeding event. 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.

Caution D.4.a in 1BwGP 100-2 warns the operator to carefully manipulate feedwater valves at low load conditions due to the greater risk of shrink and swell problems causing a reactor trip.

Feed Water Induced Steam Generator Shrink occurs when a step change in feedwater flow and/or temperature occurs. A large quantity of relatively cooler feedwater flow entering the SG downcomer causes a sudden contraction or collapse of steam bubbles in the downcomer. 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.

This effect will be worse when the recirculation ratio is smaller (i.e. lower power levels). The effect will also be worse if the difference between feedwater temperature and downcomer mixture temperature is larger (i.e. low power levels when feedwater preheating is less effective).

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.

47 **Points: 1.00**

Given:

- The control room operators are responding to a RED path condition on the heat sink status tree.
- While they attempt to restore feed flow to a S/G, conditions degrade to the point that RCS bleed-and-feed must be established.

The reason RCS bleed and feed must be established QUICKLY is to prevent...

a. thermal shock of the reactor vessel.

b. failure of the RCP seals.

c. core uncovery due to inadequate ECCS flow.

d. inadequate RCS boration due to inadequate ECCS flow.

Answer:
c.

Explanation:

The question meets the K/A, requires examinee knowledge of effect of failure in AF system (loss of heat sink) will have on the RCS.
Delays in bleed and feed initiation result in higher saturation temperatures and pressure. Analysis has shown that this can lead to inadequate safety injection flow and core uncovery.
Choice A is incorrect, thermal shock of the RCS vessel is a result of a excessive RCS cooldown event.
Choice B is incorrect, RCP seal failure is a result of a loss of all AC power event.
Choice C is correct, because rising RCS pressure may prevent sufficient ECCS flow for core cooling.
Choice D is incorrect, inadequate boration may be a problem during an ATWS, but not during a loss of heat sink.

48 **Points: 1.00**

Given:

- BOTH units are in Mode 3.
- A fault in the switchyard has caused a loss of offsite power on BOTH units.
- BOTH units ESF buses are being supplied from their respective DGs.
- NON-ESF buses 143,144 and 244 are ENERGIZED via their respective ESF to Non-ESF cross tie breakers per 1/2BwOA ELEC-4, LOSS OF OFFSITE POWER.
- Bus 243 is DE-ENERGIZED and faulted.
- A SAT fire and deluge actuation has caused an auto start of the 0A Fire Pump.
- The 0B Fire Pump did NOT auto start (but can be manually started).
- The necessary NON-ESF loads RUNNING are:
    - The 0B WS Pump.
    - The U-0 Station Air Compressor.

The EO monitoring the DGs reports the 1A DG bearing temperature is rising and the DG will need to be shutdown in the next 5 minutes.

In order to maintain the necessary NON-ESF loads WITHOUT relying on automatic equipment starts, prior to securing the 1A DG, the crew must...

a. START the 0A WS Pump and STOP the 0B WS Pump.

b. START the U-1 SAC and STOP the U-0 SAC.

c. START the 0C WS Pump and STOP the 0B WS Pump.

d. START the 0B Fire Pump and STOP the 0A Fire Pump.

Answer:
b.

Explanation:

Question meets K/A, requires examinee ability to predict impact of malfunction of AC power system and align standby equipment with its correct DG.
The following is the respective non-ESF power supplies:
0A WS pump- bus 143 (power from bus 141 and 1A DG)
0B WS pump- bus 144 (power from bus 142 and 1B DG)
0C WS pump- bus 243 (power from bus 241 and 2A DG)
U-0 SAC- bus 143 (power from bus 141 and 1A DG)
U-1 SAC- bus 144 (power from bus 142 and 1B DG)
0A Fire pump- bus 144 (power from bus 142 and 1B DG)
0B Fire pump is diesel powered (no AC source)
Choice A is incorrect, 0B WS pump is powered from bus 144.
Choice B is correct, U-0 SAC is powered from bus 143 which will be de-energized by 1A DG shutdown.
Choice C is incorrect, 0B WS pump is powered from bus 144.
Choice D is incorrect, 0A Fire pump is powered from bus 144.

49 **Points: 1.00**

Given:

- Unit 1 at 100% power, normal alignment.

The following sequence of events occurs:

- Annunciator 1-21-E10, "125V DC DIST PNL 111/113 VOLT LOW", alarms.
- DC bus 111 Voltmeter indicates 0 volts.
- The U1 Reactor is TRIPPED manually.
- The U1 Main Generator Exciter Field Breaker is locally OPENED.
- NO other actions are taken.

The following indications are now noted:

- The U1 Main Generator Output voltage indicates 0 volts.
- DC bus 111 Voltmeter (still) indicates 0 volts.

Currently, with above conditions, which of the following sets of AC buses are ENERGIZED?

a. 141, 142, 143, 144

b. 141, 143, 156, 157

c. 142, 144, 156, 159

d. 142, 144, 157, 158

Answer:
c.

Explanation:
The question meets the K/A by requiring knowledge of the effect that a loss of DC system has on the components using DC control power.
On a loss of Bus 111 (indicated by no voltage on 111/113 meter), the plant must be tripped or it will trip on low S/G levels due to the FRVs closing on the loss of DC.
DC Bus 111 supplies control power to Bus 141, but since it's already on the SAT, it will stay energized after the main generator trip. DC Bus 113 (which is also deenergized due to it being fed from 111) supplies control power to Buses 143, 157, and 159. Buses 143 and 157 are normally powered from the UAT, so they will not ABT over to the SAT on the generator trip; thus they will be deenergized. Bus 159 is already on the SAT, so it will stay 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 incorrect, see explanation above.

50 **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 the associated breaker 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.

Explanation:

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

51 **Points: 1.00**

Given:

- The Turbine Bldg Fire and Oil (F & O) sump is transferring water to the lime sludge lagoon due to unusually heavy sump input.
- The 0D (low flow) sump pump and 0A (high flow) sump pump are currently running.
- An alarm on the RM-11 indicates 0PR05J has turned RED.

As a result of the hi rad condition, the F & O sump system will...

a. trip ONLY the high flow F & O sump pump and RE-DIRECT the flowpath to the waste water treatment oil separator.

b. NOT trip the F & O sump pumps, but RE-DIRECT the flowpath to the turbine bldg floor drain tanks.

c. trip BOTH F & O sump pumps and TRIP ALL the tendon tunnel sump pumps.

d. trip BOTH F & O sump pumps and ISOLATE the flowpath to the lagoon.

Answer:
d.

Explanation:

The question meets the K/A, requires examinee knowledge of interlocks that will isolate releases to the environment upon a high rad signal.

The interlock on high rad of 0PR05J is to trip all the F & O sump pumps and close 0DO030 valve, which is the flowpath isolation to the lime sludge lagoon.
Another interlock of the system auto swaps the flow path from the wastewater treatment facility to the lime sludge lagoon when a high flow sump pump starts. This is not applicable to the hi rad situation but adds to the plausibility of distractors.
Choice A is incorrect, but plausible since another rad monitor interlock re-directs flow (SG blowdown) additionally, since the wastewater treatment facility is only designed to handle low flow input from the F & O sump, it would be logical that the high flow sump would have to be tripped to re-direct flow there.
Choice B is incorrect, but plausible since another rad monitor interlock re-directs the flow (SG blowdown) to tanks in the turbine bldg.
Choice C is incorrect, but plausible since another rad monitor (CP sump) trips input to the sump (tendon tunnel sumps are input to the F & O sump).
Choice D is correct, see explanation above.

52 **Points: 1.00**

Which of the following signals is REQUIRED to cause _SX169A/B, DG _A/B SX VLV, to AUTOMATICALLY OPEN?

a. DG speed signal.

b. ESF bus undervoltage.

c. Safety Injection signal.

d. DG output breaker closed.

Answer:
a.

Explanation:
The question meets the K/A, requires examinee knowledge of cause/effect relationship between DG and Service Water System.
The _SX169A/B valves open based upon a speed signal (>280 rpm) from the DG only.

Choice A is correct, see explanation above
Choice B is incorrect, but plausible because an ESF bus undervoltage is an auto start signal for the DGs.
Choice C is incorrect, but plausible because other SX valves reposition on an SI signal.
Choice D is incorrect, but plausible because other ESF functions are interlocked with the DG output breakers.

53 **Points: 1.00**

Which of the following Remote Shutdown Panel (RSP) events would cause MCR annunciator 1-1-C7, REMOTE S/D PANEL TROUBLE, to alarm?

a. An annunciator alarming on the RSP.

b. Loss of instrument air pressure to the RSP.

c. Placing a Remote/Local switch in "Local" at the RSP.

d. Tripping a Reactor Coolant Pump at the RSP.

Answer:
b.

Explanation:

The question meets the K/A, requires examinee ability to monitor operation of the instrument air system (at the RSP).

Choice A is incorrect, almost every other local panel in the plant has a local annunciator as an input to its MCR trouble alarm, however the RSP does not contain local annunciators.
Choice B is correct, a drop in IA pressure to <78 psig at the RSP will bring in the trouble alarm. This is important due to the air controllers that are on the RSP.
Choice C is incorrect, placing remote/local switches in local at the RSP cause alarms in the MCR for the individual equipment placed in local, but not the RSP trouble alarm.
Choice D is incorrect, RCP trip pushbuttons at the RSP are unique because they are always active (do not require taking local equipment control) however they do not bring in the RSP trouble alarm.

54 **Points: 1.00**

Given:

- Unit 1 was at 100% power, normal alignment.
- An inadvertent Phase A Containment Isolation occurs due to a SINGLE ESF relay actuation on ONE train.
- TEN MINUTES later, the crew is attempting to diagnose which train actuated.
- NO other Phase A Containment Isolation valves have been manipulated.

Which of the following CLOSED valves would POSITIVELY confirm that the relay was a TRAIN B actuation?

a. 1CV8112, SEAL WTR RTRN CNMT ISOL VLV

b. 1CV8149C, 75 GPM LTDWN ORIF 1C ISOL VLV

c. 1CC685, CC FROM RC PUMPS THERM BARR ISOL VLV

d. 1SD005B, S/G 1D BLWDN SAMPLE ISOL VLV

Answer:
a.

Explanation:
The question meets the K/A, requires examinee ability monitor ESF slave relay actuation from the MCR.

Choice A is correct, 1CV8112 is a motor operated valve that is operated by B train relay actuation only because it is on a containment penetration that has two isolation valves. Additionally 1BwOA PRI-13 specifically uses this valve to determine which train actuated.
Choice B is incorrect, although 1CV8149C receives a B train containment isolation signal (CIS), because the valve is inside containment, either train will isolate instrument air to containment and the valve will fail closed. Therefore it is not positive indication after ten minutes of a B train actuation.
Choice C is incorrect, although 1CC685 receives a phase B (NOT phase A, train B) CIS. Therefore it is not positive indication after ten minutes of a B train actuation.
Choice D is incorrect, 1SD005B is a single valve isolation of a containment penetration that receives a CIS signal from both trains.

55 **Points: 1.00**

Given:

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

The following indications are observed on the Unit 2 RCFC Dry Bulb temperatures:

- 2A RCFC Inlet Temperature 119° F.
- 2B RCFC Inlet Temperature 118° F.
- 2C RCFC Inlet Temperature 127° F.
- 2D RCFC Inlet Temperature 121° F.

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

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

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

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

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

Answer:
c.

Explanation:
The question meets the K/A, requires examinee ability to monitor system parameters to prevent exceeding design limits (Technical Specifications).
1BwOSR 0.1-1,2,3, Modes 1,2,3 Shiftly Daily Operating Surv. step F.4 describes the method for calculating containment temperature for tech spec limit comparison. This method is to calculate the average of the inlet temperatures on the running RCFCs.
Choice A is incorrect because the shut down RCFC is not calculated into the average.
Choice B is incorrect because only one RCFC over the limit does not require TS entry.
Choice C is correct, see explanation above.
Choice D is incorrect because 2D RCFC is above the limit.

56 **Points: 1.00**

Given:

- Unit 1 is at 50% power, steady state, normal alignment.

The following event occurs:

- A stationary gripper fuse blows and one shutdown bank rod drops to the bottom of the core.
- Five minutes later, the unit has NOT tripped and all automatic functions operated as designed.

Given the above condition, with NO operator action, what happened to DEMAND on 1FK-121, CENT CHG PMPS FLOW CONT VLV controller?

DEMAND...

a. initially ROSE and stabilized approx. 20% HIGHER than before the rod drop.

b. initially ROSE, then DROPPED and stabilized approx. 20% LOWER than before the rod drop.

c. remained approximately the SAME as before the rod drop.

d. initially DROPPED and stabilized approx. 20% LOWER than before the rod drop.

Answer:
c.

Explanation:
Question meets KA - question requires examinee knowledge of the effect a malfunction in the rod control system will have on CVCS.
A dropped rod will have the immediate effect of lowering Tave. However, Tref will remain relatively constant due to turbine load remaining constant. Although the lower Tave will drop steam pressure and impulse pressure slightly, the turbine control system will open governor valves to maintain impulse pressure approx. the same, thereby maintaining Tref approx. the same. Since Tref remains constant. The rod control system will automatically withdraw control rods to maintain Tave approx. the same as Tref. This will act to minimize the pressurizer level change due to Tave changes. Since Tave determines pressurizer program level, program level change will be very small and actual level will tend to follow program level since the primary is a constant mass system (changes in Tave that effect actual level will also affect program level in the same manner. The combined effect of the system automatic action is that program level and actual pressurizer level change very little, therefore 1CV121 demand changes very little. The distractors change of 20% demand after stabilizing does not happen.

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

57 **Points: 1.00**

Given the following sequence of events:

- Unit 1 has experienced a RCS LOCA combined with a loss of off site power.
- CETC Temperatures are 500 °F.
- RCS Pressure is 1200 psig.
- Pressurizer level is 30% and STABLE.
- 1A CV pump is re-aligned for normal charging at 100 gpm.
- 1B CV pump is shutdown.
- 1A SI pump flow is 400 gpm.
- 1B SI pump is tripped.

Subsequently, during performance of 1BwEP ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION, the RCS was depressurized to 800 psig over a 5 minute period. During the depressurization the following parameters changed:

- 1A SI pump flow steadily rose to 650 gpm.
- Pressurizer level rapidly rose to 80%.

Which of the following explains the amount of the rise in pressurizer level over the five minute depressurization?

a. The total amount of level rise is normal for the rise in SI flow.

b. The pressurizer level transmitters are affected by the RCS pressure drop.

c. The SI Accumulators injected during the RCS depressurization.

d. The level rise is partially due to a steam void in the reactor vessel head.

Answer:
d.

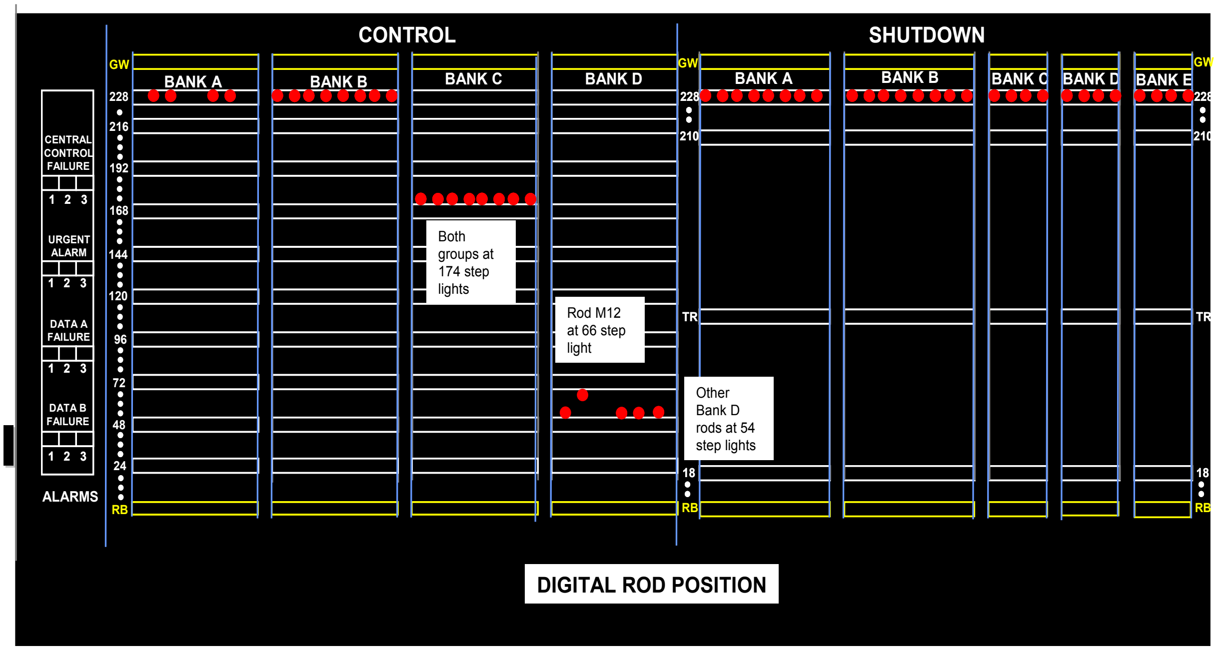
Explanation:
The question meets the K/A, requires examinee knowledge of operational implications as it applies to PZR level indication when the RCS reaches saturation conditions.

During depressurization the pressurizer level is expected to rise due to the rise in SI flow combined with a drop in break flow. However, two notes in 1BwEP ES-1.2 warn about rapidly rising PZR level due to upper head region voiding with no RCPs running to provide head region bypass cooling flow. Since the pressurizer volume is approx. 128 gal/% level, a rise of 50% would represent approx. 5760 additional gallons. The rise in SI flow of 250 gpm would only represent approx. 1250 gal. Even if break flow (originally 500 gpm, SI flow + normal charging) is considered dropping to zero, that would only add an additional 2500 gallons in five minutes.
Choice A is incorrect, see explanation above, the level rise is more than can be accounted for by rise in SI flow combined with drop in break flow.
Choice B is incorrect, pressurizer level transmitters work off the principle of DP based upon calibration at a certain temperature. A drop in pressure should not affect the accuracy of the transmitters.
Choice C is incorrect, SI accumulator injection is not expected until approx. 650 psig.
Choice D is correct, see explanation above.

58 **Points: 1.00**

Given:

- A DEHC malfunction causes a turbine load drop on unit 1 from 100% power.
- After the unit stabilizes, power is at 58%.
- The NSO then notes the below DRPI and Step Counter indications.



With the above indications, what are the conditions of the following Tech Specs?

	TS 3.1.6 CONTROL BANK INSERTION LIMITS	TS 3.1.7 ROD POSITION INDICATION
a.	limits EXCEEDED	limits EXCEEDED
b.	limits NOT exceeded	limits EXCEEDED
c.	limits NOT exceeded	limits NOT exceeded
d.	limits EXCEEDED	limits NOT exceeded

Answer:
d.

Explanation:
The question meets the K/A, requires examinee ability to monitor changes in control board rod position indication to prevent exceeding design limits.

At 58% power the rod insertion limit is bank CBD at 64 steps. This is obtained from interpolating control bank inserting limits of CBD at 0 step (30% power) and CBD at 161 steps (100% power). Current indications have both the step counters and group 2 of CBD below 64 steps.
The limit for rod group alignment is $\pm$ 12 steps (DRPI from step counters). Neither CBD DRPI group indication is > 12 steps from demanded position.

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 **Points: 1.00**

Given:

- Unit 1 is in Mode 2, with a reactor startup is in progress.
- Reactor power is approx. 500 CPS in the source range.

Then:

- Source Range channel N32 fails to zero.
- Annunciator 1-10-B1, SR HIGH VOLT FAILURE, alarms.

Which ONE of the following describes the technical specification required action for the N32 failure?

a. Open the reactor trip breakers immediately.

b. Stop positive reactivity additions immediately.

c. Verify P-6 is in the required state within 1 hour.

d. Place the failed channel in trip within 1 hour.

Answer:
b.

Explanation:
Question meets K/A, requires examinee ability to apply tech specs for the NI system.
Per tech spec 3.3.1, condition H, with one SR channel inoperable, required action is to suspend operations involving positive reactivity additions immediately.
Choice A is incorrect, opening the reactor trip breakers is required if two SR channels are failed.
Choice B is correct, see explanation above.
Choice C is incorrect, verifying the P-6 interlock is in its required state, is for failure of the P-6 interlock, however it is associated with the Intermediate Range NIs.
Choice D is incorrect, placing channel in trip within an hour is applicable to other reactor trip instrumentation that protects from a DNB accident (low PZR press, low RCS flow, RCP bus UV, etc).

60 **Points: 1.00**

Given:

- Unit has experienced a loss of core cooling event.
- The STA is monitoring the status trees for critical safety function status.

When in normal display mode, the Core Exit Thermocouple displays on 1PM05J are...
(assume all thermocouples are operable)

a. an AVERAGE of the TEN HIGHEST READING OPERABLE thermocouples in each train.

b. an AVERAGE of ALL OPERABLE thermocouples in each train.

c. the AUCTIONEERED HIGHEST reading thermocouple in each train.

d. the MEAN VALUE of ALL OPERABLE thermocouples in each train.

Answer:
a.

Explanation:
Question meets K/A, requires examinee ability to monitor CETCs during inadequate core cooling.
The CETC normal display is the average of the 10 highest CETCs.
Choice A is correct, see explanation above.
Choice B is incorrect, average of all input temperatures are used in RCS Tave circuits.
Choice C is incorrect, auctioneered high temperature is used in rod control circuit.
Choice D is incorrect, mean temperature (median select) is used in turbine control displays.

61 **Points: 1.00**

Given:

- The 0AR039J, FUEL HANDLING BLDG CRANE, area rad monitor has failed HIGH.

The above condition will prevent...

a. NEW fuel transfer from the spent fuel pool to the fuel transfer cart.

b. SPENT fuel moves within the spent fuel pool.

c. SPENT fuel transfer from the fuel transfer cart to the spent fuel pool.

d. NEW fuel transfer from shipping containers to the spent fuel pool.

Answer:
d.

Explanation:

Question meets K/A, requires examinee knowledge of how a malfunction of a rad monitoring system will effect the fuel handling equipment.

0AR039J failing high will prevent upward motion of the fuel handling bldg crane hoist. This crane is only used for new fuel transfer from shipping containers to the pool or transferring spent fuel already in casks. All the distractors can be performed by the spent fuel pool bridge crane which is not affected by the failure.

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.

62 **Points: 1.00**

Given:

- Unit 1 is at 28% reactor power in a normal lineup.
- Steam Dump Mode Select switch is in the TAVE position.
- Subsequently, the main turbine trips.

With the above conditions, and NO operator action, the steam dumps will maintain RCS temperature at ...

a. 550 °F.

b. 557 °F.

c. 560 °F.

d. 561 °F.

Answer:c.

Explanation:

The question meets the K/A, requires examinee knowledge of the loss of load bistable upon a turbine loss of load.

When the turbine trips below 30% reactor power, the reactor will not trip due to the C-8 interlock. The steam dumps will be armed by C-7 and the load reject bistables and controller will maintain Tave at 560°F because of the 3°F deadband programed in to the load reject controller.

Choice A is incorrect, 550°F is the setpoint for P-12, the steam dump block interlock.

Choice B is incorrect, 557°F is the setpoint for the plant trip controller.

Choice C is correct, see explanation above.

Choice D is incorrect, 561°F is the setpoint for the SG PORVs.

63 **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.

Explanation:

The question meets the K/A, requires examinee ability to monitor waste gas system after a rad monitor high alarm.

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.

64 **Points: 1.00**

Given:

- A tornado damaged the Braidwood Switchyard causing a loss of all AC power to both Units' SAT's.
- The 2B DG is the ONLY emergency diesel that started and automatically loaded.
- NONE of the other DG's can be started.

With the above conditions, Bus 242 will...

a. be crosstied to Bus 142. The 2B SX pump is the ONLY SX pump that will be run, with Unit 1 and Unit 2 SX systems crosstied.

b. NOT be crosstied to Bus 142. The 2B SX pump is the ONLY SX pump that will be run, with Unit 1 and Unit 2 SX systems NOT crosstied.

c. be crosstied to Bus 142. BOTH the 1B and 2B SX pumps will be run.

d. NOT be crosstied to Bus 142. The 2B SX pump is the ONLY SX pump that will be run, with Unit 1 and Unit 2 SX systems crosstied.

Answer:
a.

Explanation:
The question meets the K/A because the candidate must know the power supply alignment to the SX pumps during a limited AC crosstie.
Per 1BwCA 0.0, bus 142 and 242 will be crosstied. Therefore the 2B DG will supply both busses 142 and 242, but 1BwCA 0.0 will not allow a single DG to power (2) SX pumps due to load restrictions, so only 1 SX pump (2B) will supply SX to both units via a SX system unit crosstie.
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.

65 **ID: RS20079-N01** **Points: 1.00**

Prior to starting a Station Air Compressor per BwOP SA-1, STARTUP AND OPERATION OF STATION AIR COMPRESSORS, minimum instrument air pressure is required to...

a. position the compressor inlet, unloader and discharge valves.

b. meet a compressor minimum control air starting interlock.

c. clear a compressor low air pressure trip signal.

d. position the compressor cooling water supply valves.

Answer:
a.

Explanation:
The question meets the K/A, requires knowledge of the cause-effect relationship between the IA system and the SA system compressor.
Per BwOP SA-1, a minimum of 50 psig is required from instrument air to start a SAC. This minimum pressure is to provide enough control air to align the SAC flow control valves (inlet, unloader and disch) to pressurize the SA and IA system. Without proper control air the disch valve fails closed and the unloader valve fails open. Therefore the SAC will not pressurize the system without control air.
Choice A is correct, see explanation above.
Choice B is incorrect, compressor does not have starting interlock for control air pressure.
Choice C is incorrect, compressor does not have trip for control air pressure.
Choice D is incorrect, although the compressor have high oil and air temperature trips, unlike other turbine bldg equipment, the cooling water supply valves are not controlled by instrument air.

66 **Points: 1.00**

Given:

- The U-1 Unit Supervisor is currently behind the control boards attending a brief.
- Subsequently a transient occurs on U-1 requiring an immediate operator action step from an approved procedure to stabilize the plant.

In accordance with OP-AA-101-111, ROLES AND RESPONSIBILITIES OF ON-SHIFT PERSONNEL, while performing the immediate action the Reactor Operator will...

a. wait until an SRO is at the Unit desk and agrees with verbalization of the action before performing it.

b. perform the action without delay from memory even if the US is NOT available to confirm the actions.

c. wait until the U-1 admin NSO peer checks the action before performing it.

d. wait until the procedure is in hand and they have read the action before performing it.

Answer:
d.

Explanation:
The question meets the K/A, requires examinee ability to manage crew actions during plant transients. Section 4.7.2 part 1 of OP-AA-101-111 states it is the RO's responsibility to perform immediate operator actions of an abnormal procedure from memory. It is also preceded by a note that says the immediate actions to stabilize the plant during transients take priority over verbalization of the action to the US.
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.

67 **Points: 1.00**

Given:

- An operating surveillance is commenced on a Monday morning that is expected to take several days to complete.
- On Wednesday afternoon of the same week, a RO assigned to continue work on the surveillance discovers that the procedure revision being used was superseded just that Wednesday morning.

In accordance with HU-AA-104-101, PROCEDURE USE AND ADHERENCE, the completion of the surveillance...

a. can be completed using the superseded (old) revision. Since that revision was current when the surveillance was started, NO supervisor evaluation is required.

b. data and placekeeping MUST be transferred to the latest revision. The superseded revision CANNOT be used.

c. can be completed using the superseded (old) revision. However, a temporary procedure change must be made to the old revision to match the current revision.

d. can be completed using the superseded (old) revision. Additionally, a supervisor MUST evaluate that NO fatal flaws exist with the old revision.

Answer:
d.

Explanation:
The question meets the K/A, requires examinee ability to perform plant procedures during all modes of operation (i.e. under different situations as they arise)
Section 3.1.3 of HU-AA-104-101 states that only current revisions of procedures can be used but then qualifies the statement to include superseded revisions for applicable work provided a supervisor review is performed to ensure no fatal flaws exist.
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 **Points: 1.00**

Given:

- A xenon free reactor startup is in progress on Unit 1 using control rods only.
- The reactor is exactly critical in the intermediate range below P-6.
- IR range SUR is 0 DPM.
- Control Bank D (CBD) control rods are at 135 steps.

The US then directs the RO to establish a steady state Startup Rate of 0.5 DPM, raise power to approx. 10^{-8} amps, then level power for recording of critical data.

- The RO withdraws CBD rods 10 steps and establishes the 0.5 DPM steady startup rate.

To raise power to 10^{-8} amps and then maintain power stable, the RO will...

a. maintain CBD at 145 steps. When power reaches approx. 10^{-8} amps, negative reactivity from ITC will level power.

b. periodically withdraw CBD as power rises to maintain the steady 0.5 DPM Startup Rate. When power reaches approx. 10^{-8} amps, insert CBD to approx. 145 steps to maintain power steady.

c. maintain CBD at 145 steps. When power reaches approx. 10^{-8} amps, insert CBD to approx. 125 steps to maintain power steady.

d. maintain CBD at 145 steps. When power reaches approx. 10^{-8} amps, insert CBD to approx. 135 steps to maintain power steady.

Answer:
d.

Explanation:

The question meets the K/A, requires examinee ability to manipulate console controls (rod control) as required between shutdown conditions and desired power level.

With a xenon free start up in progress and the reactor exactly critical in the intermediate range, once a steady state startup rate has been established the reactor will maintain that startup rate until the point of adding heat is reached. However, 10^{-8} amps is well below the point of adding heat. Therefore the RO will have to manually stop the power rise by inserting control rods. Since the reactor was exactly critical with CBD at 135 steps prior to the power rise, the RO will need to return CBD to approx. 135 steps to maintain exactly critical conditions again and level power at 10^{-8} amps.

Choice A is incorrect, power will not level off at 10^{-8} amps like it would at reaching the POAH because there is no negative reactivity feedback from ITC at 10^{-8} amps.

Choice B is incorrect, rods will not need to be withdrawn more than 145 steps to maintain the 0.5 DPM startup rate once a steady state startup rate has been establish. Also, leaving rods at 145 steps will not steady power at 10^{-8} amps (power will continue to rise).

Choice C is incorrect, inserting rods to 125 steps after reaching 10^{-8} amps will make the reactor subcritical and establish a negative startup rate instead of maintaining power at 10^{-8} amps.

Choice D is correct, see explanation above.

69 **Points: 1.00**

Which of the following types of reactor accident categories is the OTDT reactor trip designed to protect against?

a. Start-up accident

b. Loss of Heat Sink accident

c. Departure From Nucleate Boiling accident

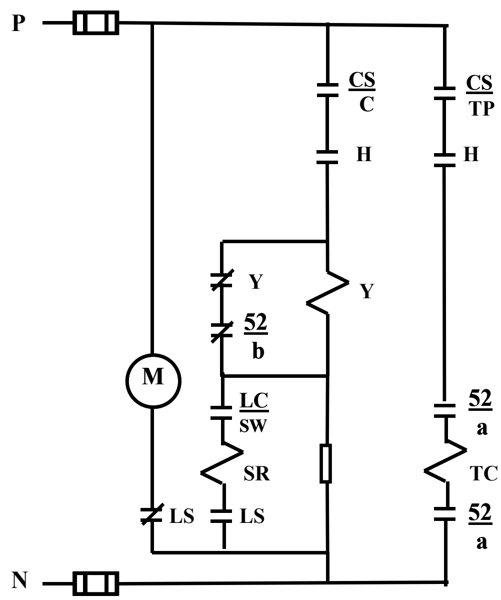
d. RCS Integrity accident

Answer:
c.

Explanation:
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.
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 terminated by either a direct low pressure reactor trip or by an OTDT reactor trip. Events raising RCS temperature are terminated by an OTDT trip and operation of SG PORVs and safety valves.
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 a RCS loop DT to assure the OTDT will "see" a slow adverse reactivity trend.
Choice D is incorrect, Hi Pressurizer Pressure trip is protection against loss of RCS pressure boundary integrity.

70 **Points: 1.00**

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



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

a. Y

b. 52/b

c. LC/SW

d. LS

Answer:
a.

Explanation:
The question meets the K/A, requires examinee ability to interpret station electrical drawings.
The circuit shown is a simplified example of the anti-pumping circuit used in all Braidwood 4KV and 6.9 KV breakers. All the contact options are in the circuit leg with the spring release (SR) coil, therefore any of the contacts can interrupt the circuit continuity to the SR coil.
Choice A is correct, the Y (anti-pumping) contact opens whenever the Y coil is energized (i.e. when the breaker has a active closing signal). The open Y contact will prevent the SR coil from getting a second closing signal (from a continuous close signal) in the event the breaker closed and tripped back open immediately. In that case the Y coil would remain energized through the resistor downstream of the Y coil, thereby keeping the Y contact open and preventing the breaker from reclosing after the closing spring had re-charged.
Choice B is incorrect, the 52/b (breaker position) contact is in the circuit as a protective device that will prevent a closing signal from energizing the SR coil on a breaker that is already closed (thus protecting the SR coil from prolonged energization and coil damage).
Choice C is incorrect, the LC/SW (latch check switch) contact is is a contact in a microswitch that senses the position of the breaker trip coil plunger, thereby preventing a closing signal from energizing the SR coil on a breaker that has an active trip signal in.
Choice D is incorrect, the LS (limit switch) contact is a limit switch that senses the charge state of the closing spring. Its function is to prevent a closing signal from energizing the SR coil on a breaker that does not have the closing spring charged and to energize the closing spring charging motor.

71 **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 MCR SRO directs an RO to monitor control room intake air for elevated radiation trends.
- The RO notes all MCR rad monitor icons on the RM-11 GRID 2, PROCESS AIR MONITORS, are currently GREEN.

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

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

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

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

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

Answer:
a.

Explanation:
The question meets the K/A, requires examinee ability to use radiation monitoring systems.
MCR rad monitor icons are green even when their sampled plenums are not online because the sample pumps will continuously sample plenums that have stagnant air flow. With the 0A VC system in normal alignment (outside air intake) the only rad monitors that would have MCR intake air flow through their respective intake plenum is the 0PR31J and 32J.

Choice A is correct, see explanation above.
Choice B is incorrect, 0PR33J and 34J sample the outside air intake from Unit 2 (0B train). Although the radiation spill was in the 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 spill 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 spill 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.

72 **Points: 1.00**

You are assigned a task in a tank room in the Aux Building. The room was recently surveyed and the following radiological conditions are listed on the RWP survey map:

- General area radiation 110 mR/Hr.
- Airborne radiation is 0.25 DAC.
- Contamination levels of 250 dpm/100 cm^2 beta/gamma and 25 dpm/cm^2 alpha.

When you arrive at the room, there are 2 posted signs as follows:
- Caution - Radiation Area
- Caution - Contaminated Area

What actions (if any) are required/allowed and why?

a. Do NOT proceed, notify RP Department because the Radiation Area posting is INCORRECT.

b. Do NOT proceed, notify RP Department because the Contaminated Area posting is INCORRECT.

c. Do NOT proceed, notify RP Department because the room also requires an additional posting of Caution - Airborne Radioactivity Area.

d. Proceed with assigned task because ALL postings are CORRECT and complete.

Answer:
a.

Explanation:
Question meets KA - question requires examinee knowledge of radiological safety principles pertaining to licensed operator duties.
Per RP-AA-376, area meets requirements for posting as high rad area (> 100mR/Hr) and contaminated area (> 20dpm/100cm2 alpha), but not for Airborne Radioactivity (< 0.3 DAC) .
A is correct, see explanation above.
B is incorrect, Contaminated Area posting is correct.
C is incorrect, no Airborne Radioactivity posting is required.
D is incorrect, Radiation Area Posting is not correct.

73 **Points: 1.00**

During a refueling outage, while moving irradiated fuel in containment, it is the ROs responsibility to maintain Refuel Cavity Water Level at $\geq$ 23' above the ...

a. bottom of the refueling cavity.

b. reactor vessel flange.

c. top of the fuel in the reactor vessel.

d. top of any fuel in transit in containment.

Answer:
b.

Explanation:
Question meets KA - question requires examinee knowledge of radiological safety principles pertaining to licensed operator duties.
Tech Spec 3.9.7 states that refueling cavity level must be maintained $\geq$23' above the reactor vessel flange. The bases for this requirement is a minimum amount of water volume to ensure the radiological consequences of a postulated fuel handling accident inside containment are within limits. Specifically, it ensures enough water to retain sufficient iodine fission product activity.
A is incorrect, although plausible assuming the refuel cavity level indication is referenced to the bottom of the cavity, however the level indication is referenced to the site reference elevation.
B is correct, see explanation above.
C is incorrect, although plausible because Tech Spec 3.7.14 requires $\geq$23' of water in the spent fuel pool above the top of the fuel assemblies in the storage racks.
D is incorrect, although plausible assuming that the water requirement is based upon shielding thickness for surface rad levels. Although the water also serves this function, the bases is actually for a total volume of water as stated above.

74 **Points: 1.00**

While transitioning to 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, after completing the actions of 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, an orange path is identified under SUBCRITICALITY.

With regard to implementing 1BwFR-S.1, RESPONSE TO NUCLEAR POWER GENERATION/ATWS, procedure usage rules require the crew to...

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

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

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

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

Answer:
a.

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

75 **Points: 1.00**

Given:

- A fire has occurred in the Main Control Room envelope.
- BOTH Units in the main control room have been evacuated.
- 0/1/2BwOA PRI-5, MAIN CONTROL ROOM INACCESSIBILITY, have been implemented.
- Both Remote Shutdown Panels (RSDP) have been activated.
- The fire brigade is preparing to purge the MCR per BwOP VC-7, PURGE OF THE CONTROL ROOM WITH 100% OUTSIDE AIR.
- The fire chief requests operators start the 0B VC supply and return fans.

The operator will start the 0B VC supply and return fans at…

a. 0VC01JB, Control Room System HVAC Local Control Panel (451' L-26).

b. 1PL04J, Division 11 RSDP (383' N-23).

c. 1PL05JA Division 12 RSDP (364' M-23).

d. 2PL04J, Division 21 RSDP (383' L-27).

Answer:
c.

Explanation:
The question meets the K/A, requires examinee knowledge of RO task performed outside the MCR during an emergency.
Original plant design included BOTH Train A and Train B VC RSDP controls at the Unit 1 RSDP. Train B VC supply and exhaust fans local controls were moved as a plant modification and operated from 1PL05JA at 364 M-23.
A is incorrect, panel is 0B VC train, but VC supply and return fans are not included at 0VC01JB.
B is incorrect, would be correct for train A VC components.
C is correct, see explanation above.
D is incorrect, Unit 0 components are operated at Unit 1 RSDP panels.

76 **Points: 1.00**

Given:

- Unit 1 experienced a stuck open pressurizer safety valve at 100% power.
- The reactor was manually tripped and SI was actuated.
- Subsequently, the safety valve reseated.
- The 1A Auxiliary Feedwater pump is out of service.

The following alarms are LIT:

- UNIT 1 AREA FIRE (0-37-A4)
- AF PUMP TRIP (1-3-A6)
- AF PUMP AUTO START (1-3-B6)
- AF PUMP SUCT PRESS LOW (1-3-A7)
- AF PUMP SX SUCT VLVS ARMED (1-3-E7)

An EO reports the 1B Auxiliary Feedwater pump suction pressure switch is damaged from an electrical short (small fire).

With the above conditions, which of the following procedures contain the specific steps that will be successful in establishing a running AF Pump?

a. 1BwOA PRI-5, CONTROL ROOM INACCESSIBILITY.

b. 1BwOA ELEC-5, LOCAL EMERGENCY CONTROL OF SAFE SHUTDOWN EQUIPMENT.

c. BwOP AF-7, AUXILIARY FEEDWATER PUMP _B STARTUP ON RECIRC.

d. 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION.

Answer:
c.

Explanation:
The question meets the K/A, requires examinee knowledge of local auxiliary operator tasks during an emergency.
The question is SRO level because it requires assessment of conditions and selecting of procedures.
With the AF pump trip alarm up, and report of suction pressure switch failure, the 1B AF pump is tripped. The low suction pressure trip can only be overridden by taking the local switch to "Start With Bypass" at the 364' Emergency Control Panel. This is directed in BwOP AF-7.
Choice A is incorrect, 1BwOA PRI-5 contains direction for starting the AF pump at the 383' Unit 1 Remote Shutdown Panel, however the pump will not start from there with the above conditions.
Choice B is incorrect, 1BwOA ELEC-5 contains direction for starting the 1A AF pump locally, but will not work for the 1B AF.
Choice C is correct, see explanation above.
Choice D is incorrect, 1BwEP-0 contains direction for manually starting the AF pps from the MCR, however the pump will not start from there with the above conditions.

77 **Points: 1.00**

Given:

- Unit 1 is in Mode 5 with RCS cooldown in progress.
- Current RCS temperature is 195°F.

Subsequently, the following occurs:

- The running RH pump trips.
- The MCR crew swapped to the standby RH train ten minutes after the pump trip.
- RCS temperature rose 10°F during the transient.

In accordance with the Exelon Reportability Manual, this event is...
(references are available)

a. NOT reportable.

b. reportable within 15 minutes (maximum).

c. reportable within 1 hour (maximum).

d. reportable within 4 hours (maximum).

Answer:
b.

Explanation:

The question meets the K/A, requires examinee knowledge of events related to system operations that must be reported to outside agencies and requires examinee determine reportability requirements.
The question is SRO level because determining event reportability is a function unique to the SRO position.
The event meets the criteria for an EAL classified MU5 (Cold Shutdown Matrix) event. Therefore, per LS-AA-1020 the maximum reporting time is 15 minutes because of the activation of the emergency plan (SAF 1.1).
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.

78 **Points: 1.00**

Given:
- Unit 1 is at 100% power, all systems normally aligned.

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

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

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

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

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

d. trip the reactor, actuate SI and enter 1BwEP-0, REACTOR TRIP OR SAFETY INJECTION, because there is a secondary steam break in containment.

Answer:
d.

Explanation:
The question meets the K/A, requires examinee ability to interpret steam line rupture conditions requiring an ESFAS actuation.
The question is SRO level because it requires assessment of conditions and selection of procedures.
Choice A is incorrect, a RCS break would not result in rising reactor power or lowering RCS temperature.
Choice B is incorrect, although 1BwOA PRI-16 entry conditions are met, the event is not a dilution because dilutions would raise RCS temperature instead of lowering it. Also dilution would not cause severe PZR pressure and level changes.
Choice C is incorrect, a 3% PZR level drop in one minute would easily exceed the capacity of a charging pump, therefore even if this was an RCS leak, PRI-1 would not be appropriate.
Choice D is correct, steam break will account for all indications given and an SI is required because a steam break inside containment will eventually lead to an SI based upon steamline low pressure or containment high pressure.

79 **Points: 1.00**

Given:

- Unit 1 was at 100% power, all systems normally aligned.
- An RCS LOCA occurred.
- 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, has been implemented.

The crew is performing 1BwEP-1, step 12, "Check if RCS Cooldown and Depressurization is Required" with the following conditions:

- RWST level is 62% and slowly lowering.
- RCS pressure is 275 psig.
- 1A & 1B RH pump flows (1FI-618/619) indicate 0 gpm.

Based on the above conditions, the NEXT procedure the Unit Supervisor will direct the crew to implement is…

a. 1BwCA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION.

b. 1BwCA-1.3, SUMP BLOCKAGE CONTROL ROOM GUIDELINE.

c. 1BwEP ES-1.2, POST LOCA COOLDOWN AND DEPRESSURIZATION.

d. 1BwEP ES-1.3, TRANSFER TO COLD LEG RECIRCULATION.

Answer:
c.

Explanation:

Question meets KA - question requires examinee determine conditions necessary for recovery from the accident and transition to a procedure for recovery actions during a large break LOCA (1BwEP ES-1.2).
The question is SRO level because it requires assessment of conditions and selection of procedures.
A is incorrect, 1BwCA-1.1 is entered following alignment to cold leg recirculation. With stated RWST level, ECCS is still aligned to RWST. Also, 1BwCA-1.1 is entered if no RH recirc train is available. No indication in stem that RH train not available.
B is incorrect, 1BwCA-1.3 is entered following alignment to cold leg recirculation. With stated RWST level, ECCS is still aligned to RWST.
C is correct, 1BwEP-1 directs transition to 1BwEP ES-1.2 with RCS press < 325 psig and RH pump flow < 1000 gpm.
D is incorrect, 1BwEP ES-1.3 would not be implemented until RWST level < 46%.

80 **Points: 1.00**

Given:

- Unit 1 was at 100% power, all systems normally aligned.
- A gradual loss of instrument air pressure occurs and the crew is performing 0/1BwOA SEC-4, LOSS OF INSTRUMENT AIR.
- When air pressure drops to reactor trip criteria, the crew trips the reactor and enters 1BwEP-0, REACTOR TRIP AND SI.
- When the reactor trips, an inadvertent SI also occurs.

Which ONE (1) of the following describes the usage of 1BwOA SEC-4 while responding to this event?

a. Discontinue use of 1BwOA SEC-4 (emergency procedures contain steps for restoring instrument air).

b. Continue with 1BwOA SEC-4 in conjunction with 1BwEP-0 ONLY after immediate actions are complete.

c. Continue with 1BwOA SEC-4 ONLY after transitioning out of 1BwEP-0.

d. Continue with 1BwOA SEC-4 ONLY after ALL Emergency Procedures are complete.

Answer:
b.

Explanation:

Question meets KA - question requires examinee knowledge of how Abnormal Operating Procedures are used in conjunction with EOPS during a loss of instrument air event.

The question is SRO level because it requires assessment of conditions and selection of (when to use) procedures.

1BwOA SEC-4, step 3.c. directs the procedure to be performed in conjunction with 1BwEP-0, however, immediate actions steps take precedent over this because they are verifying the status of key safety functions. Emergency procedures do not assume a loss of instrument air, therefore the crew should be attempting to regain instrument air in conjunction with the emergency procedures as soon as the immediate action steps are complete.

A is incorrect, emergency procedures do not assume a loss of instrument air, therefore many steps will need instrument air to properly perform them.

B is correct, see explanation above.

C is incorrect, 1BwOA SEC-4, step 3.c. directs the procedure to be performed in conjunction with 1BwEP-0.

D is incorrect, 1BwOA SEC-4, step 3.c. directs the procedure to be performed in conjunction with 1BwEP-0.

81 **Points: 1.00**

Given:
- During an RCS LOCA on Unit 1, emergency coolant recirculation capability was lost, and 1BwCA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is currently in progress.
- Subsequently, the STA identifies a containment condition requiring transition to BwFR-Z.1, RESPONSE TO HIGH CONTAINMENT PRESSURE.

When the crew performs 1BwFR-Z.1, the containment spray pumps will be operated per...

a. 1BwCA-1.1, because it ALWAYS provides for REDUCED containment spray regardless of containment conditions.

b. 1BwCA-1.1, because it MAY provide for REDUCED containment spray depending upon containment conditions (but NOT always).

c. 1BwFR-Z.1, because it provides for MAXIMUM containment spray to restore the safety function.

d. 1BwFR-Z.1, because functional restoration procedures take precedent over contingency actions.

Answer:
b.

Explanation:
Question meets KA - question requires examinee ability to select appropriate procedure during emergency operations involving a loss of emergency coolant recirc.
The question is SRO level because it requires assessment of conditions and selection of procedures.
A note prior to step 3 in 1BwFR-Z.1 says that if 1BwCA-1.1 is in effect, CS should be operated as directed in 1BwCA-1.1. This particular note would be specific to the SRO function (that the SRO would "own") vs. other EOP notes which would typically be assigned to an RO to monitor. The background document for 1BwFR-Z.1 explains that the note is to provide for a possibly less restrictive CS usage in order to conserve RWST water. The table in 1BwCA-1.1 directs either 0, 1 or 2 CS pumps to be run depending upon the containment conditions (containment pressure and number of RCFCs in operation), while 1BwFR-Z.1 always directs maximum available CS operation. Therefore, the CS requirement in 1BwCA-1.1 MAY be less restrictive, but not always. The candidate does not have to have the CS usage table in 1BwCA-1.1 memorized to answer the question, they only must know that there may or may not be conditions to support reduced CS flow. While some notes in procedures are specific to RO actions, this note is unique to the SRO function as it will be his responsibility to direct the mitigation strategy.
A is incorrect, 1BwCA-1.1 does not always reduce containment spray If containment pressure is >20 psig and <2 RCFCs are in operation, or if containment pressure is >50 psig, 2 CS pumps are still required.
B is correct, see explanation above.
C is incorrect, 1BwFR-Z.1 is not correct.
D is incorrect, 1BwFR-Z.1 is not correct.

82 **Points: 1.00**

Given:

- An accidental gaseous release is occurring in the Aux. Bldg.
- The Shift Emergency Director has directed you to determine the station total release rate for the EAL evaluation.

The following plant conditions exist on Units 1 & 2:

- Unit 1 & 2 Wide Range Gas Monitors - 1/2PR030J are INOPERABLE.
- Aux Building Supply Fan - 0VA01CA is running.
- Aux Building Exhaust Fan - 0VA02CA is running.
- ALL other Aux Building Supply and Exhaust Fans are stopped.
- Unit 1 Lab Exhaust Fan - 0VL02CA is running.
- Service Building and Solid Radwaste Fan - 0VW03CA is running.
- Aux Building Filtered Vent Fan - 0VF01CA is running.
- Unit 1 Vent Stack Effluent Flow - 1VA019 is INOPERABLE.
- Unit 2 Vent Stack Effluent Flow - 2VA020 is 8.65×10^{3} CFM.
- Unit 1 Vent Stack Effluent High Range Gas (1PD428) is 7.4×10^{-3} microCi/cc (highest reading channel).
- Unit 2 Vent Stack Effluent Low Range Gas (2PB128) is 9.5×10^{-4} microCi/cc (highest reading channel).

The station total release rate is approximately...
(references are available)

a. 1.59×10^{3} microCi/sec..

b. 4.76×10^{5} microCi/sec.

c. 7.20×10^{5} microCi/sec.

d. 7.51×10^{5} microCi/sec.

Answer:
d.

Explanation:
Question meets KA - question requires examinee ability to perform procedure during emergency operations accidental gaseous release.
The question is SRO level because determining release rates for EAL classification is an SRO function.

Unit 1 release rate:
$7.4E^{-3}$ (given) x $2.14E^{5}$ (sum of estimated VA & VL fans flow) x 472 (conversion factor) =747459
Unit 2 release rate:
$9.5E^{-4}$ (given) x $8.60E^{3}$ (given) x 472 (conversion factor) = 3856

Total release rate:
747459 + 3856= 751315 or $7.51E^{5}$

A is incorrect, this is result if 472 cc/sec/CFM multiplier is not used.
B is incorrect, this is result if wrong VA fan estimated number is used (2 exhaust fans running vs. only 1).
C is incorrect, this is result if VL fan estimated flow is omitted from calculation.
D is correct, see explanation above.

83 **Points: 1.00**

Given:

- Spent fuel moves are in progress in the Spent Fuel Pool.
- The RO monitoring the RM-11 reports that an ALERT condition has just alarmed on rad monitor 0AR063J, FUEL HANDLING BLDG GENERAL.

The above meets entry conditions for, and the SRO will direct entry into...

(Procedure names listed for reference)

1) 1BwOA REFUEL-1, FUEL HANDLING EMERGENCY
2) 1BwOA REFUEL-2, FUELING CAVITY OR SPENT FUEL POOL LEVEL LOSS
3) 0BwOA REFUEL-3, LOSS OF SPENT FUEL POOL COOLING

a. 1 & 2, but NOT 3.

b. 2 & 3, but NOT 1.

c. 1 & 3, but NOT 2.

d. 1, 2 & 3

Answer:
a.

Explanation:
Question meets KA - question requires examinee ability to recognize abnormal conditions that are entry level conditions for abnormal operating procedures.
The question is SRO level because it requires assessment of conditions and selection of procedures. Additionally, the question is SRO level based upon the guidance of section II.D. (second bullet) of "Clarification Guidance for SRO-Only Questions" (rev. 1) (from NRC website) and 10CFR55.43(b)(4) link that this is a question based upon radiation hazards that may arise during normal and abnormal situations and the analysis and interpretation of radiation readings as they pertain to selection of abnormal procedures.
Generically, Fuel Handling Bldg. radiation levels rising are listed as entry conditions for 1BwOA REFUEL-1 and 1BwOA REFUEL-2, but not 0BwOA REFUEL-3. Although the specific rad monitor in the question stem is not listed as an entry condition to the listed abnormal procedures (other abnormal procedures such as OA PRI-4 entry conditions list a specific rad monitor), the SRO duty would be to analyze the indications as such. Additionally BwAR 4-0AR063J, refers to 1(2)BwOA REFUEL-1 & 1(2)BwOA REFUEL-2.
A is correct, see explanation above.
B is incorrect, see explanation above.
C is incorrect, see explanation above.
D is incorrect, see explanation above.

84 **Points: 1.00**

Given:

- A Unit 1 reactor trip and SI occurred from full power due to lowering pressurizer pressure.
- During step 3 of 1BwEP-0, REACTOR TRIP OR SI, the crew noted BOTH ESF buses de-energized and transitioned to 1BwCA 0.0, LOSS OF ALL AC.
- 30 minutes later, one ESF bus has been restored, and the required actions of 1BwCA-0.2, LOSS OF ALL AC RECOVERY WITH SI REQUIRED, have been completed.
- The crew is ready to transition to the next required procedure.

The following plant conditions exist:

- NO CV OR SI pumps are running or available.
- Containment pressure is 2.5 psig and slowly rising.
- RCS pressure is 500 psig and slowly rising.
- SG pressures are ALL 1115 psig and STABLE.
- CETC temperature is 1250 °F and slowly rising.

With the above conditions, the US will direct a procedure transition to...

a. 1BwEP-0, step 3. Complete the 1BwEP-0 immediate actions, then transition to 1BwFR-C.1, RESPOND TO INADEQUATE CORE COOLING.

b. 1BwEP-0, step 3. Complete the actions of 1BwEP-0 until transition is directed to 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT.

c. 1BwFR-C.1, RESPOND TO INADEQUATE CORE COOLING, and complete the actions of 1BwFR-C.1.

d. 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT, and complete the actions of 1BwEP-1.

Answer:
c.

Explanation:

Question meets KA - question requires examinee ability to recognize the difference between a LOCA and a inadequate core cooling event.
The question is SRO level because it requires assessment of conditions and selection of procedures.
The event described will require a transition to 1BwFR-C.1 due to a status tree red path on core cooling (CETC >1200°F). Status trees were not implemented prior to this point because 1BwCA-0.0 and 0.2 both take precedent over the BwFRs.
Following completion of 1BwCA-0.2 actions, a note exist just prior to step 12 that directs BwFR to be implemented as necessary. Therefore the next transition should be 1BwFR-C.1. This particular note would be specific to the SRO function (that the SRO would "own") vs. other EOP notes which would typically be assigned to an RO to monitor.
A is incorrect, return to 1BwEP-0 is not required. Although return to "procedure and step in effect" is common transition during emergency procedure use, it does not apply in the situation.
B is incorrect, return to 1BwEP-0 is not required. Although return to "procedure and step in effect" is common transition during emergency procedure use, it does not apply in the situation.
C is correct, see explanation above.
D is incorrect, although 1BwCA-0.2 directs transition to 1BwEP-1, performing steps in 1BwEP-1 would not be appropriate with a red path in core cooling.

85 **Points: 1.00**

Given:

- Unit 1 is at full power.
- Letdown flow is currently 75 gpm.
- The other two (2) letdown orifice isolation valves are CLOSED, but available.
- All other equipment is normally aligned for this power level.
- Elevated RCS coolant activity is reported by chemistry and the crew enters 1BwOA PRI-4, HIGH REACTOR COOLANT ACTIVITY and Tech Spec 3.4.16, RCS SPECIFIC ACTIVITY.

(1) As part of the mitigation strategy of 1BwOA PRI-4, the US will direct the crew to ___________________.
(2) The bases for the LCO 3.4.16 activity limits is to prevent exceeding dose acceptance criteria during a(n) ____________________.

a. (1) MAXIMIZE letdown per BwOP CV-9, LETDOWN ORIFICE OPERATION
(2) SGTR or MSLB

b. (1) MAXIMIZE letdown per BwOP CV-9, LETDOWN ORIFICE OPERATION
(2) RCS LOCA outside containment

c. (1) ISOLATE letdown per BwOP CV-17, ESTABLISHING AND SECURING NORMAL AND RH LETDOWN FLOW
(2) RCS LOCA outside containment

d. (1) ISOLATE letdown per BwOP CV-17, ESTABLISHING AND SECURING NORMAL AND RH LETDOWN FLOW
(2) SGTR or MSLB

Answer:
a.

Explanation:
Question meets KA - question requires examinee ability to determine corrective actions associated with high RCS activity.
The question is SRO level because it requires assessment of conditions and selection of procedures. Also, the question is SRO level because it requires knowledge of Tech Spec bases.
The mitigation strategy of 1BwOA PRI-4 includes maximizing letdown to clean up the RCS with a mixed bed demin. The bases for Tech Spec 3.4.16 specifically list the accidents of SGTR and MSLB as events the LCO limits are intended to prevent exceeding dose limits during.
A is correct, see explanation above.
B is incorrect, see explanation above.
C is incorrect, see explanation above.
D is incorrect, see explanation above.

86 **Points: 1.00**

Given:

- Unit 1 is at 100% power.
- PRT level and pressure are currently 80% and 2.0 psig.
- On line RHUT level is currently 75%.
- The 0A and 0B Waste Gas Compressors are OOS.
- ALL other equipment is normally aligned.

The following then occurs:

- 0GW9293, GDT TO RHUT PCV, fails OPEN causing the Waste Gas Vent Header pressure to rise to 8.0 psig.

With the above conditions, PRT pressure will _______(1)______, and after 0GW9293 is isolated, the Waste Gas Vent Header pressure can be adjusted by lowering the _________________(2)________________.

a. (1) remain at 2.0 psig
   (2) PRT level per BwOP RY-4, DRAINING THE PRESSURIZER RELIEF TANK

b. (1) remain at 2.0 psig
   (2) on line RHUT level per BwOP AB-12, RECYCLE HOLD UP TANK OPERATIONS

c. (1) rise to 6.0 psig
   (2) on line RHUT level per BwOP AB-12, RECYCLE HOLD UP TANK OPERATIONS

d. (1) rise to 8.0 psig
   (2) PRT level per BwOP RY-4, DRAINING THE PRESSURIZER RELIEF TANK

Answer:
b.

Explanation:

Question meets KA - question requires examinee ability to predict the impact on the PRT of overpressurizing the waste gas header and select a procedure to mitigate the overpressurization.
The question is SRO level because it requires assessment of conditions and selection of procedures.
Over pressurizing the waste gas (GW) vent header will not affect PRT pressure during normal operations because the PRT is normally isolated from the GW vent header by a normally closed air operated valve (1RY469). Additionally, the normal gas supply to the PRT is nitrogen, therefore lowering PRT level will cause nitrogen gas to flow into the PRT but will not lower GW vent header pressure. Without the Waste Gas Compressors available, another viable method to lower vent header pressure is to lower RHUT level (GW vent header provides cover gas to RHUTS, therefore lowering RHUT level would allow vent header pressure to drop.
A is incorrect, lower PRT level will not affect vent header pressure.
B is correct, see explanation above.
C is incorrect, PRT pressure will not rise, however this is plausible because 1RY469 has an interlock that auto closes the isolation valve at 6.0 psig in the PRT.
D is incorrect, PRT pressure will not rise.

87 **Points: 1.00**

Given:

- Unit 1 is at 20% power.
- All systems are normally aligned.
- PZR pressure is 2205 psig and lowering.
- 1RY455C, PZR Spray Valve, is open and NOT responding in manual OR auto.
- All other PZR system components are operating as designed.

Based on the above indications, the Unit Supervisor will direct the crew to…

a. manually trip the reactor, enter 1BwEP-0, REACTOR TRIP OR SI, and STOP the 1C RCP.

b. manually trip the reactor, enter 1BwEP-0, REACTOR TRIP OR SI, and STOP the 1D RCP.

c. STOP the 1C RCP per BwOP RC-2, SHUTDOWN OF A RCP and verify all PZR heaters energized.

d. STOP the 1D RCP per BwOP RC-2, SHUTDOWN OF A RCP and verify all PZR heaters energized.

Answer:
a.

Explanation:
Question meets KA- question requires examinee to determine PZR spray valve has malfunctioned and determine procedure requirements for mitigating consequences of spray valve failure.
The question is SRO level because it requires assessment of conditions and selection of procedures.
Spray valve 1RY455C has failed open. 1BwOA INST-2 directs manually tripping reactor and tripping associated RCP.
A is correct, procedurally directed response.
B is incorrect, 1D RCP supplies PZR spray valve 1RY455B.
C is incorrect, operator must manually trip reactor prior to tripping 1C RCP because it is not within Braidwood accident analysis to operate the reactor critical with less than 4 RCPs running.
D is incorrect, operator must manually trip reactor prior to tripping RCP and stated RCP supplies 1RY455D.

88 **Points: 1.00**

Given:

- Unit 1 was at 100% power.
- During the previous shift, Power Range Channel N41 failed LOW.
- ALL the actions of 1BwOA INST-1 were completed for channel N41.

Subsequently, Power Range Channel N44 fails HIGH.

15 minutes later, with the above conditions, to re-energize Source Range N31 & N32 instruments, the US will direct the crew to…

a. REMOVE the N41 control power fuses at 1PM07J per 1BwOA INST-1, NUCLEAR INSTRUMENTATION MALFUNCTION.

b. BYPASS channel N41 at 1PM07J MISC CONTROL AND INDICATION PANEL per 1BwOA INST-1, NUCLEAR INSTRUMENTATION MALFUNCTION.

c. INSTALL jumpers in the SSPS input bays per 1BwOA PRI-5, CONTROL ROOM INACCESSIBILITY.

d. momentarily PLACE the SR MAN BLOCK switches in RESET at 1PM05J per 1BwEP ES-0.1.

Answer:
c.

Explanation:
Question meets KA- question requires examinee knowledge of RO tasks performed outside the MCR and the resultant effect.
The question is SRO level because it requires assessment of conditions and selection of procedures.

When actions were taken for N41, control power fuses are removed and channel fails in conservative state (>P-10). Two PRNI channels that are de-energized or failed high will prevent the SR channels from re-energizing because P-10 will not reset (requires 3 of 4 channels below 10%).
A is incorrect, removing control power fuses will not reset P-10 because bistables fail in conservative state (>P-10).
B is incorrect, bypassing control functions will defeat the power mismatch alarms and rod stops, but not affect P-10.
C is correct, the only way to reset P-10 in this state is to jumper the relays in SSPS per 1BwOA PRI-5.
D is incorrect, the SR block/reset switches will bypass the P-6 auto re-energization interlock, but not the P-10 interlock.

89 **Points: 1.00**

Given:

- Unit 1 is at 25% reactor power.
- The 1B FW pump is RUNNING.
- The 1C FW pump is SHUTDOWN.
- ALL other plant equipment is normally aligned for this power level.

The following then occurs:

- Annunciator 1-16-B1, FW PUMP 1B TRIP alarms on 1PM04J.

With the above conditions, the Unit Supervisor will direct the crew to…

a. START the 1A FW PUMP per 1BwOA SEC-1, SECONDARY PUMP TRIP.

b. START the STANDBY CD/CB PUMP per 1BwOA SEC-1, SECONDARY PUMP TRIP.

c. TRIP the TURBINE and enter 1BwOA TG-8, TURBINE TRIP BELOW P-8.

d. TRIP the REACTOR and enter 1BwEP-0, REACTOR TRIP OR SI.

Answer:
d.

Explanation:

Question meets KA- question requires examinee to interpret control room indications and verify system status and make proper directive for system conditions.

The question is SRO level because it requires assessment of conditions and selection of procedures. Also it requires knowledge of the content of 1BwOA SEC-1 versus the mitigation strategy.

Per 1BwOA SEC-1, if no FW pumps are running, then the reactor must be tripped.

A is incorrect, although starting the standby 1A FW pump at this power level may prevent an automatic trip of the plant, it is not procedurally allowed.

B is incorrect, starting the standby CD/CB pump will not produce enough forward feed flow to feed the S/Gs at current steam pressure.

C is incorrect, although tripping the turbine at less than 30% power is an option for a turbine trip condition, it is not allowed as this situation (lo-2 S/G level) would cause an auto reactor trip.

D is correct, see explanation above.

90 **Points: 1.00**

Given:

- Unit 2 is at 50% power.
- The 2B AF Pump was shutdown 30 minutes ago following a monthly surveillance run.
- The EO performing field operations reports that the 2B AF Pump discharge piping is 150°F and has risen since the pump was secured.

With the above conditions, the US will enter ___________(1)___________ and the subsequent procedural recovery steps ______(2)_______ require the 2B AF pump to be declared INOPERABLE.

a. (1) 2BwOA SEC-7, AUXILIARY FEEDWATER CHECK VALVE LEAKAGE
(2) WILL

b. (1) 2BwOA SEC-6, LOSS OF FEEDWATER TEMPERING LINE SUBCOOLING
(2) WILL

c. (1) 2BwOA SEC-7, AUXILIARY FEEDWATER CHECK VALVE LEAKAGE
(2) will NOT

d. (1) 2BwOA SEC-6, LOSS OF FEEDWATER TEMPERING LINE SUBCOOLING
(2) will NOT

Answer:
a.

Explanation:
Question meets KA- question requires examinee to predict the impact of MFW backleakage into the AF system and use procedures to mitigate the consequences.
The question is SRO level because it requires assessment of conditions and selection of procedures. Additionally it requires knowledge of the content of the procedure concerning the operability status of safety related equipment.
Entry conditions for 2BwOA SEC-7 is any AF line temperature greater than 130°F. The AF piping temperature will rise if back leakage through check valves from the MFW system is occurring. The major concern is temperature in the AF system rising to the level of steam binding and inoperability of an AF pump. The most plausible distractors for this question are FW tempering flow nozzle shock and 2BwOA SEC-6. This concern and procedure will be entered when low flow is detected in the tempering line which the AF header combines with just prior to entering the S/Gs.
With AF discharge line temperature at 150°F, 2BwOA SEC-7 is going to require the respective 2AF013_ valves to be closed and the AF piping to cool. Closing the 2AF013_ valves makes the AF pp inoperable as it will not automatically be able to perform it's safety function.
A is correct, see explanation above.
B is incorrect, see explanation above.
C is incorrect, see explanation above.
D is incorrect, see explanation above.

91 **Points: 1.00**

Given:

- Unit 1 is in an outage and maintenance activities are ongoing in containment.
- Containment integrity is being maintained.
- Operations is preparing a containment release package to perform a "bleed and feed" of containment with the Mini-Purge System to improve working conditions for personnel.
- Current containment pressure is + 0.5 psig.
- The US is preparing to brief an NSO on performing the release per BwOP VQ-6, CONTAINMENT MINI-PURGE SYSTEM OPERATION, by marking up the appropriate procedure steps.

With the above conditions, the US will direct the NSO to NOT start the 1VQ04C, MINI-PURGE SUPPLY FAN,...

a. until containment integrity is NO longer required.

b. because containment pressure is currently too LOW.

c. because containment pressure is currently too HIGH.

d. until the mini-purge exhaust fan is running to satisfy an electrical interlock.

Answer:
c.

Explanation:

Question meets KA- question requires examinee to explain and apply system precautions.
The question is SRO level because it involves a procedure that performs a radioactive release to the environment. RP-BR-980 specifically instructs the SRO to "instruct the operator to perform the release per the applicable procedure (BwOP VQ-6). BwOP VQ-6 contains independent steps for operating the mini-purge system during extended purge ("feed and bleed") operations. Also in BwOP VQ-6 is limitation E.8 and notes prior to starting mini-purge supply fan to ensure containment pressure is < 0.3 psig to ensure forward flow. Past OPEX has shown running fans without forward flow has damaged fan blading. Although, SROs routinely review and mark up operating procedures for briefing the operators, this question is SRO level because the radioactive release content makes it unique to the SRO position.
A is incorrect, maintaining containment integrity does not prevent starting mini-flow purge fan as long as the containment isolation function of the dampers is operational.
B is incorrect, see explanation above.
C is correct, see explanation above.
D is incorrect, although fan interlocks exist in other systems, not for mini-flow purge system.

92 **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 NOT lit.
- Annunciator 1-10-B6, ROD BANK LOW INSERTION LIMIT, is LIT.

  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, the US will direct the crew to...

a. borate the RCS per BwOP CV-6, because boration is required by 1BwOA SEC-1, and boration is required by Tech Specs.

b. EMERGENCY borate the RCS per 1BwOA PRI-2, because EMERGENCY boration is required by 1BwOA SEC-1, and boration is required by Tech Specs.

c. borate the RCS per BwOP CV-6, because boration is required by 1BwOA SEC-1, but NOT required by Tech Specs.

d. EMERGENCY borate the RCS per 1BwOA PRI-2, because boration is required by 1BwOA SEC-1, but NOT required by Tech Specs.

Answer:
c.

Explanation:
Question meets KA- question requires examinee to evaluate plant conditions and use procedure to correct condition
The question is SRO level because it requires assessment of facility conditions and selection of proper procedure.
With control rods below the rod insertion limit (RIL) LOW alarm and above the RIL LO-2 alarm, 1BwOA SEC-1 requires borating the RCS, but Tech Specs does not (since tech spec RIL has not been violated until LO-2 alarm is lit. However, neither situation is 1BwOA PRI-2 entry criteria for emergency boration.
A is incorrect, boration not required by tech specs (RIL not exceeded).
B is incorrect, 1BwOA PRI-2 entry conditions not met.
C is correct, see explanation above.
D is incorrect, 1BwOA PRI-2 entry conditions not met.

93 **Points: 1.00**

Given:

- Unit 1 is in MODE 3 during a normal shutdown and cooldown.
- RCS Pressure is currently 1200 psig.

A few minutes ago, a Containment Area Rad Monitor alarmed and the crew noted the following:

- Letdown is ISOLATED.
- PZR Level is STABLE.
- Charging flow is 150 gpm.

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

a. 1BwEP-1, LOSS OF REACTOR OR SECONDARY COOLANT

b. 1BwOA PRI-1, EXCESSIVE PRIMARY PLANT LEAKAGE

c. 1BwOA PRI-4, HIGH REACTOR COOLANT ACTIVITY

d. 1BwOA S/D-2, SHUTDOWN LOCA

Answer:
b.

Explanation:

The question meets the K/A, requires examinee knowledge of abnormal condition procedures.
The question is SRO level because it requires assessment of conditions and selection of appropriate procedure. Additionally, the question is SRO level based upon the guidance of section II.D. (second bullet) of "Clarification Guidance for SRO-Only Questions" (rev. 1) (from NRC website) and 10CFR55.43(b)(4) link that this is a question based upon radiation hazards that may arise during normal and abnormal situations and the analysis and interpretation of radiation readings as they pertain to selection of abnormal procedures.
Choice A is incorrect, 1BwEP-1 would not be the next procedure entry because it is not a direct entry procedure.
Choice B is correct, 1BwOA PRI-1 is written for leaks in Mode 1-3 with SI accumulators not isolated.
Choice C is incorrect, 1BwOA PRI-4 is applicable for the gross failed fuel monitor alarm, not a containment area monitor.
Choice D is incorrect, 1BwOA S/D 2 is written for modes 3 and 4 after SI accumulators are isolated. Mitigation steps of 1BwOA S/D-2 will be to manually restore and align ECCS, then cool down the plant and depressurize.

94 **Points: 1.00**

Per BwAP 340-1, USE OF PROCEDURES FOR OPERATING DEPARTMENT, which of the following correctly describes when an emergency procedure action on the Operator Action Summary (OAS) page, is applicable?

a. Only PRIOR to performing the applicable step in the main body of the procedure.

b. Only after proceeding PAST the applicable step in the main body of the procedure, BUT it will NEVER apply after a transition is made to another procedure.

c. Only after proceeding PAST the applicable step in the main body of the procedure, AND it MAY apply after a transition is made to another procedure.

d. ANY time during the applicable procedure performance, unless a specific procedural starting point is referenced in the action.

Answer:
d.

Explanation:

The question meets the K/A, requires examinee ability to interpret and execute procedure steps. Examinee must know rules of usage for operator action summary actions in order to properly execute them.

The question is SRO level because SROs read and direct the emergency procedure actions and have the responsibility of monitoring the OAS actions as the procedure is being performed. Although ROs are responsible for the content of the OAS, the SRO is responsible for knowing the rules of usage and when to implement OAS actions.

BwAP 340-1 states that a step on the OAS page contains information that must be monitored throughout the procedure.

Choice A is incorrect, OAS actions apply as soon as the procedure is entered.

Choice B is incorrect, however, this may be applicable to continuous action summary steps depending on the action and whether it is superseded or no longer applicable to the next procedure.

Choice C is incorrect, however, this may be applicable to continuous action summary steps depending on the action and whether it is superseded or no longer applicable to the next procedure.

Choice D is correct, see explanation above.

95 **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 LOCA load demand.

b. 7 days of post LOCA load demand.

c. 14 days of post LOOP shutdown load demand.

d. 30 days of post LOOP shutdown load demand.

Answer:
b.

Explanation:
The question meets the K/A, requires examinee ability to explain system limits.
Tech Spec 3.8.3 bases states the DG are supplied with enough stored oil for 7 days of post LOCA loads.
The question is SRO level because it requires knowledge of Tech Spec bases.
Choice 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.

96 **Points: 1.00**

Given the following separate plant conditions and contingency actions, which one will the US REFUSE to allow?

a. The U-1 main generator oil vapor extractor trips. To maintain generator loop seal tank vapors below explosive limits, a portable air mover is connected to the vapor extractor flame arrester.

b. A DG is running loaded during during a loss of off site power event, when the DG ventilation supply damper failed closed due to inadvertent HELB actuation. The damper was then manually re-opened and secured with tie-wraps.

c. The 1A DG #1 air compressor is OOS. To maintain the associated DG air receiver at normal pressure, a hose is used to temporarily cross tie that DGs two air receivers and raise pressure back to normal on the #1 receiver.

d. An ESF Battery Room Exhaust Fan trips. To maintain the ESF battery room temperature within Tech Spec limits, a portable electric fan is placed in the doorway to exhaust air from the battery room.

Answer:
d.

Explanation:

The question meets the K/A, requires examinee knowledge of process for making operating changes to the facility.
The question is SRO level because it is the SRO responsibility to assess facility conditions and select procedures to mitigate plant malfunctions.
All of the operations have an associated approved procedure for performing the continency actions except placing an electrical fan in the doorway of an ESF battery room. This condition is an OPEX (lesson learned) from Braidwood station during the late 1980s. 1BwOS DC-2-1a, AAR 125 VDC BATTERY ROOM VENTILATION does give an option for installing temporary alternate ventilation to a battery room, but specifically says it must be NON-electrical.
Choice A is incorrect, BwAR 1-18-C12 provides actions for this contingency.
Choice B is incorrect, BwOP VD-6 provides actions for this contingency.
Choice C is incorrect, BwOP DG-24 provides actions for this contingency.
Choice D is correct, see explanation above.

97 **Points: 1.00**

Tech Spec 3.4.6 prohibits starting an RCP if any RCS cold leg temperature is less than or equal to 350°F when any S/G secondary water temperature is greater than or equal to 50°F above each of the RCS cold leg temperatures.

What is the basis for this restriction?

a. Prevent outsurge from emptying the pressurizer following an RCP start.

b. Minimize RCS pressure transient caused by reverse heat transfer from a hot S/G.

c. Minimize RCS pressure transient caused by additional heat transfer from the core.

d. Minimize RCS pressure transient due to additional RCP pressure head added to RCS pressure.

Answer:
b.

Explanation:

Question meets KA - question requires examinee knowledge of technical specifications limiting conditions for operations bases.
The question is SRO level because it requires knowledge of Tech Spec bases.
LCO 3.4.6 bases places restriction on starting an RCP to prevent a low temperature overpressure event due to the thermal transient when an RCP is started. The heat added by the SGs will expand the RCS, causing a pressure rise.
A is incorrect, a higher S/G temperature will cause an insurge to the Pzr, due to expansion of the coolant.
B is correct, see explanation above.
C is incorrect, the RCS pressure transient will not be caused by heat transfer out of the core. It will be into the core.
D is incorrect, RCP pressure head while increased will not cause a large pressure transient. The RCP head pressure will only be felt on the U-Tubes of the S/G.

98 **Points: 1.00**

Given:

- Unit 1 is mode 2, reactor start up in progress.
- An emergent activity requires a containment entry.

In accordance with BwAP 1450-1, ACCESS TO CONTAINMENT, which one of the following Unit 1 activities must the US ensure does NOT occur during the containment entry?

a. A mode change to Mode 1.

b. A mode change to Mode 3.

c. A containment release.

d. An RCFC fan swap.

Answer:
a.

Explanation:
The question meets the K/A, requires examinee knowledge of radiological safety procedures pertaining to containment entry.
The question is SRO level because mode changes are authorized by the SM and require SRO concurrence.
BwAP 1450-1, Access To Containment, step F.3.b.1) restricts operations from changing modes to a higher power level with personnel in containment. However, changing modes to a lower power level is acceptable.
Choice A is correct, see explanation above.
Choice B is incorrect, see explanation above.
Choice C is incorrect, a containment release does not have any restrictions during containment entry.
Choice D is incorrect, a RCFC swap does not have any restrictions during containment entry.

99 **Points: 1.00**

The crew is responding to a RED PATH on the Heat Sink status tree and has implemented 1BwFR-H.1 "RESPONSE TO LOSS OF SECONDARY HEAT SINK" when the STA identifies a RED PATH on the Core Cooling status tree.

Under these conditions, the US will ...

a. transition to 1BwFR-C.1 "RESPONSE TO INADEQUATE CORE COOLING," and will NOT direct any additional 1BwFR-H.1 actions with 1BwFR-C.1 in progress.

b. transition to 1BwFR-C.1 "RESPONSE TO INADEQUATE CORE COOLING," and concurrently direct 1BwFR-H.1 actions as time permits.

c. continue with 1BwFR-H.1, and will NOT direct any 1BwFR-C.1 "RESPONSE TO INADEQUATE CORE COOLING" actions with 1BwFR-H.1 in progress.

d. continue with 1BwFR-H.1, and concurrently direct 1BwFR-C.1 "RESPONSE TO INADEQUATE CORE COOLING" actions as time permits.

Answer:
a.

Explanation:
The question meets the K/A, requires examinee knowledge of logic used to assess safety status function. The question is SRO level because it requires assessment of conditions and selection of appropriate procedure. It is the function of the STA to report the results of the status tree pass to the SRO and then the SRO responsibility to assess and implement those results.
Per BwAP 340-1, If a red path of higher priority is encountered (Core Cooling is higher priority than Heat Sink) then the lower priority BwFR should be suspended, and the higher level red path BwFR implemented.
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.

100 **Points: 1.00**

Given:

- The crew is currently staffed with the minimum number of required qualified fire brigade members per BwAP 320-1, SHIFT STAFFING.
- Six hours into a twelve hour shift, a fire brigade member must leave work unexpectedly.

Under these conditions, to meet the requirements of BwAP 320-1, the SM/designee...

a. does NOT need to call out a replacement BECAUSE the Fire Brigade is allowed one unexpected absence for a PARTIAL shift.

b. does NOT need to call out a replacement BECAUSE the Fire Brigade Chief can fill the member's role during an unexpected absence for a PARTIAL shift.

c. must take action WITHIN 2 hours to call out a replacement AND have the position filled as soon as possible after that.

d. must take IMMEDIATE action to call out a replacement AND have the position filled within 2 hours.

Answer:
d.

Explanation:
The question meets the K/A, requires examinee knowledge of fire brigade requirements.
The question is SRO level because maintaining minimum shift staffing is the responsibility of the SM (SRO) and is normally delegated to the WEC supervisor during emergent backshift situations.
Per BwAP 320-1, step C.2, Fire Brigade composition may be less than the minimum requirements for a period not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required position.
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.