Name: _____

HLT 4 NRC Exam
Form: 0
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1. 201001K5.02 001/2/2/CRD/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Following a reactor scram in which the Control Rod Drive (CRD) pump remains running, the 2C11-R600 "CRD Flow Controller" flow INDICATION indicates significantly __(1)__ than normal.

The implication of this flow indication is that __(2)__ will occur if the reactor scram is NOT reset.

A. (1) higher

(2) excessive reactor vessel bottom head cooldown

B. (1) higher

(2) CRD pump runout

C. (1) lower

(2) excessive reactor vessel bottom head cooldown

D. (1) lower

(2) CRD pump runout

Link to KA: Tests whether the candidate understands how CRD flow indication is affected by a scram, and then asks the implications of this flow rate indication.

A. **Correct**
The charging water header taps off downstream of the CRD flow element (upstream of the CRD flow control valves, 2C11-F002A/B). When the SDV and the reactor pressures are equalized with the scram **not** reset, the charging water header will add to RPV inventory utilizing the cooling water flow path and leaking past the CRDM seals. This cooler water will cool the bottom head area and will eventually result in violation of TS cooldown rate limits.

B. **Incorrect**
1st part is correct (see "A" description)
2nd part is incorrect. **Plausible** since high flowrates are indications of pump runout. Pump runout is only prevented by use of orifices and a throttled valve in the HCU accumulator charging water header.

C. **Incorrect**
1st part is incorrect. **Plausible** since this response requires the candidate to realize which side of 2C11-F002 the flow element is on. Up stream side (with charging water header downstream of the flow element) results in a high flow indication, downstream side would result in a low flow indication.
2nd part is correct (see "A" description)

D. **Incorrect**
1st part is incorrect. (see "C" description).
2nd part is incorrect. **Plausible** because the candidate must distinguish between actual flow and indicated flow, then determine whether actual flow and indicated flow are the same. If the candidate assumes the flow element is downstream of 2C11-F002 and the charging header upstream, then a low flow indication would be expected with an actual high flow condition in the charging water header.

SYSTEM: 201001 Control Rod Drive Hydraulic System

K5. Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD DRIVEHYDRAULIC SYSTEM : (CFR: : : 41.5 / 45.3)

K5.02 Flow indication . . . . . RO 2.6 / SRO 2.6

Reference(s) used to develop this question:

C11-CRD-LP-00101 Control Rod Drive System lesson plan (section II.a.d, and Fig 01)
B31-RRS-LP-00401, "Reactor Recirc Lesson" (section III.K.4)
34AB-C71-001-2, "Scram Procedure"
Obj 18 (EO 001.013.a.07)

Reference(s) provided to the student:
None

2. 202001K4.07 001/2/2/RECIRC/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 is operating at 80% power.

- o Both reactor recirculation pumps are running at 78% speed
- o All Condensate and Condensate Booster Pumps are in service

- 4160 VAC Bus "1C" is inadvertently de-energized.

Which ONE of the following describes how the Reactor Recirculation system will be affected?

A.✓ Both Reactor Recirc pumps will automatically trip.

B. Both Reactor Recirc pumps will be running at 61% speed.

C. Both Reactor Recirc pumps will be running at 78% speed.

D. The "1A" Reactor Recirc pump will automatically trip.

The "1B" Reactor Recirc pump will be running at 61% speed.

Description of event.
A loss of 4160 "1C" will result in a loss of the Reactor Recirc AC oil pumps which causes a loss of oil pressure to the Recirc MG sets. This results in a trip of BOTH Recirc MG Sets. A unit difference exists, U2 has Adjusttable Speed Drives (ASDs) installed, U1 uses Recirc MG sets. The "1C" Condensate and "1C" Condensate Booster Pumps (CBP) will lose power when this bus is lost.

A. **Correct;** see description above.

B. **Incorrect;** see description above.
Plausible since low condensate discharge pressure will result in a #3 speed limiter. It makes sense that when a condensate pump is lost ("1C" 4160 VAC Bus supplies power to the "1C" Condensate Pump and the "1C" CBP at rated power, low CBP suction pressure will occur, which results a #3 runback signal. If the candidate does not remember that a loss of "1C" 4160 VAC bus results in a loss of the Recirc MG set AC oil pumps, this will appear to be the correct answer.

C. **Incorrect;** see description above.
Plausible since 2 Condensate and 2 CBPs are adequate to maintain RPV water level at this reactor power. The candidate may focus on the loss of the "1C" Condensate Pump and the "1C" CBP (which will actually be lost) without considering the impact of a loss of oil to the Recirc MG sets.

D. **Incorrect;** see description above
Plausible if the candidate thinks the AC oil pumps have independent power supplies and that the loss of the Condensate pumps will result in a runback on the remaining Recirc Pump.

SYSTEM: 202001 Recirculation System

K4. Knowledge of RECIRCULATION System design feature(s) and/or interlocks which provide for the following:(CFR: 41.7)

K4.07 Motor generator set trips: Plant-Specific . . . . . RO 2.8 / SRO 2.9

Reference(s) used to develop this question:

34AB-R23-004-1, Loss of 4160V bus 1A, 1B, 1C, or 1D
34AB-R23-002-1, Loss of 600V bus 1A, 1AA, 1BB, or 1B
34AR-602-101-1, Drive Motor A Trip
B31-RRS-LP-00401, Reactor Recirculation System lesson plan

Reference(s) provided to the student:
None

3. 203000G2.4.8 001/2/1/RHR/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 was operating at 100% power when a Loss Of Coolant Accident occurred.

These conditions exist:

- o The reactor has scrammed and all control rods fully inserted
- o RPV pressure ..... 960 psig, increasing at 4 psig/minute
- o Drywell (DW) pressure ... 7 psig
- o BOTH RHR loops..... DW spray mode
- o RPV level ..... -145 inches, decreasing at 2"/minute
- o 4160 VAC bus "2E" has de-energized and cannot be re-energized

IAW 34AB-R23-001-2, "Loss of 600 Volt Emergency Bus", energizing 600 VAC bus "2C" using the 4160/600V "2CD" Transformer is _____.

A. NOT allowed until the reactor is in Mode 4

B. allowed since it supports the Emergency Operating Procedures

C. NOT allowed because the "1B" Emergency Diesel Generator would be overloaded

D. allowed, but is NOT required since all low pressure Emergency Core Cooling Systems will inject at rated flow, when required

Description: The CD Transformer is only allowed to be used when in Mode 4 or 5 **OR** as required to support the EOPs. "1B" EDG loading must be considered when the "1B EDG is supplying the bus.
In this case, the CD transformer is needed to support the EOPs. D/W spray valves must be closed to ensure all ECCS water is injected to the RPV rather than being diverted to the DW. The "2A" RHR loop will not inject at rated flow to the reactor because the DW spray valves are de-energized in the open position. This will allow water to be diverted from the core to the D/W.
In this case, the "2F" bus is being supplied from offsite power rather than the "1B" EDG, so the "1B" EDG is not running and therefore, not loaded so the consideration (at this time) of diesel loading is not an issue.

A. **Incorrect;** see description above.
Plausible since Mode 4 is a minimum requirement when not in the EOPs.

B. **Correct;** see description above.

C. **Incorrect;** see description above.
Plausible since the "1B" EDG loading is a limiting criteria for use of the CD Transformer. If the "1B" EDG were running (instead of the "1A" EDG) the candidate would have to take EDG loading into the decision making process for use of the CD Transformer.

D. **Incorrect;** see description above.
Plausible since the "2C" RHR pump will run and the injection valve will open; however, water will be diverted to the DW (see description above)

SYSTEM: 203000 RHR/LPCI: Injection Mode (Plant Specific)

2.4 Emergency Procedures / Plan

2.4.8 Knowledge of how abnormal operating procedures are used in conjunction with EOPs. (CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 3.8/ SRO 4.5

Reference(s) used to develop this question:

CP-1, Alternate Level Control, Steam Cooling & Emergency RPV Depress EOP flowchart
DI-OPS-59, Operations management Expectations
34AB-R23-001-2 "Loss of 600 Volt Emergency Bus" (Pg 2 of 7, step 4.3)
34SO-E11-010-02, "RHR System" Attach 2, RHR system electrical lineup

Reference(s) provided to the student:
None

4. 204000A4.09 001/2/2/RWCU/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 2 is at 2% power performing a reactor heatup and pressurization.

The Reactor Water Cleanup (RWCU) system is aligned for vessel level control (Blowdown mode to the Main Condenser).

- o Reactor Water Level (RWL) begins to increase
- o Non-Regenerative Heat Exchanger (NRHX) Effluent temperature ..... 133°F

Which ONE of the following completes both of these statements?

NRHX Effluent temperature is monitored at 2H11-___(1)___.

IAW 34SO-G31-003-2, "Reactor Water Cleanup System", 2G31-R606, Manual Controller for 2G31-F033, "RWCU Blowdown Flow Control Valve" is required to be throttled ___(2)___.

A. (1) P602

(2) open to lower RWL

B. (1) P602

(2) closed to lower NRHX effluent temperature

C. (1) P614

(2) open to lower RWL

D. (1) P614

(2) closed to lower NRHX effluent temperature

Description
The RWCU temperatures are monitored at the 2H11-P602 panel. 34SO-G31-001-2 directs the operator to throttle 2G31-F033 to maintain Non-Regenerative Heat Exchanger (NRHX) Effluent temperature below 125°F, and 34AR-602-427-2 "RWCU FILTER INLET TEMP HIGH" requires dump flow be reduced to clear the high temp alarm (130°F). The operator is using dump flow to maintain RWL during the RPV heatup. As level goes up, normally the dump flow is raised to maintain RWL within a band. In this case the temperature limits are being exceeded and dump flow must be reduced

A. **Incorrect;** see description above
1st part is correct
2nd part is incorrect since procedure temperature limits are being exceeded. **Plausible** since RWCU is currently lined up for dump flow and RPV level is increasing.

B. **Correct;** see description above

C. **Incorrect;** see description above
1st part is incorrect; **plausible** since 2H11-P614 contains 5 digital multipoint temperature recorders and 2 SCC temperature chart recorders (many temperature points recorded), including RWCU Room Temperatures, Bottom head drain (606 recorder).
2nd part is incorrect since procedure temperature limits are being exceeded. Plausible since RWCU is currently lined up for dump flow and RPV level is increasing.

D. **Incorrect;** see description above
1st part is incorrect; **plausible** since 2H11-P614 contains 5 digital multipoint temperature recorders and 2 SCC temperature chart recorders (many temperature points recorded), including RWCU Room Temperatures, Bottom head drain (606 recorder).
2nd part is correct; see description above

KA:

SYSTEM: 204000 Reactor Water Cleanup System

A4. Ability to manually operate and/or monitor in the control room:
(CFR: 41.7 / 45.5 to 45.8)

A4.09 Reactor water temperature . . . . . RO 2.9 / SRO 2.9

Reference(s) used to develop this question:

34AR-602-427-2, "RWCU Filter Inlet Temp High" Annunciator Response

Reference(s) provided to the student:
None

5. 205000K2.01 001/2/1/RHR/BANK MOD/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 is in Mode 4 with "1C" and "1D" Residual Heat Removal (RHR) pumps running in Shutdown Cooling.

- An over-current condition develops on Emergency Bus 4160VAC "1F" which causes it to de-energize.

Which ONE of the following indicates the status of Shutdown Cooling?

A. "1C" RHR Pump has stopped

"1D" RHR Pump has stopped

B. "1C" RHR Pump is running

"1D" RHR Pump has stopped

C. "1C" RHR Pump has stopped

"1D" RHR Pump is running

D. "1C" RHR Pump is running

"1D" RHR Pump is running

Description:
The "1F" bus supplies power to the "1C" and "1D" RHR pumps. When this bus is lost, those pumps will no longer be operating.

Plausibility:
The KA for this question requires the candidate to know the power supplies to the RHR pumps when RHR is in SDC mode of operation. Plausibility for distracters is based on whether or not the candidate actually does remember the power supply lists, and whether the candidate confuses the various pumps supplied by the buses (i.e. RHRSW pump "1D" is powered by 4160 VAC bus "1G" while RHR pump "1D" is powered from 4160 VAC bus "1F")

A. **Correct**; see description and plausibility statement above

B. **Incorrect**; see description and plausibility statement above

C. **Incorrect**; see description and plausibility statement above

D. **Incorrect**; see description and plausibility statement above

SYSTEM: 205000 Shutdown Cooling System (RHR Shutdown Cooling Mode)

K2. Knowledge of electrical power supplies to the following: (CFR: 41.7)

K2.01 Pump motors . . . . RO 3.1* /3.1*

Reference(s) used to develop this question:

E11-RHR-LP-00701, Pg 81
EO 006.001.a.02
34SO-E11-010-1, Residual Heat Removal System section (7.4.4)

Reference(s) provided to the student:
None

6. 206000K5.08 001/2/1/HPCI VACUUM BREAKER/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 100% power.

- o 34SV-E41-002-2, "HPCI PUMP OPERABILITY" has just been completed.
- o When the High Pressure Coolant Injection (HPCI) system is secured, BOTH HPCI Turbine Exhaust Vacuum Breaker check valves (2E41-F102 and F103) fail in the OPEN position

- If the HPCI system receives an automatic start signal, HPCI will ___(1)___ and ___(2)___.

A. (1) experience a significant water hammer event in its exhaust line

(2) this event can be diagnosed from control room indications

B.✓ (1) discharge steam directly to the Suppression Chamber air space

(2) this steam discharge can be isolated by closing 2E41-F104

C. (1) experience a significant water hammer event in its exhaust line

(2) this event can NOT be diagnosed from control room indications

D. (1) discharge steam directly to the Suppression Chamber air space

(2) this steam discharge can ONLY be isolated by tripping HPCI

Description of this condition
The HPCI exhaust line vacuum breakers are installed to prevent siphoning water into the HPCI exhaust line. The line connects to the Torus air space. Both check valves failed in the open position will allow steam to pass through the line directly into the suppression chamber air space. The HPCI exhaust line has two motor operated isolation valves (2E41-F104 and F111) which can be operated from the Main Control Room (P601/P602).

Plausibility - Water hammer will occur if these valves failed in the closed position rather than the open position. This requires the candidate to demonstrate understanding of how the Vacuum Breakers function. If the candidates reverse their thinking about how this vacuum breaker line functions, and considering that there are no indications of the vacuum breaker check valves in the Main Control Room, some candidates may/will assume that the event can NOT be diagnosed from there. Others may assume that indication of exhaust pressure fluctuations or torus pressure increase will give evidence of the water hammer event.

A. **Incorrect**; see description and plausibility statement above

B. **Correct**; see description above

C. **Incorrect**; see description and plausibility statement above

D. **Incorrect**; see description and plausibility statement above

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SYSTEM: 206000 High Pressure Coolant Injection System

K5. Knowledge of the operational implications of the following concepts as they apply to HIGH PRESSURE COOLANT INJECTION SYSTEM : (CFR: 41.5 / 45.3)

K5.08 Vacuum breaker operation: BWR-2,3,4 . . . . . RO 3.0 / SRO 3.2

Reference(s) used to develop this question:

E41-HPCI-LP-00501, HPCI system lesson plan/simplified one-line diagram.
EO 005.001.a.04

Reference(s) provided to the student:
None

7. 206000K6.02 001/2/1/HPCI/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 100%.

- o 2R24-S022, "250 VDC RX BLDG ESSEN. 2B" de-energizes (remains de-energized)
- o A transient results in a scram and a High Pressure Coolant Injection (HPCI) automatic initiation signal.

Which ONE of the following describes how the HPCI system will respond to the automatic initiation signal?

HPCI will _____.

A. automatically start and operate normally

B. automatically start; however, only local control will be possible

C. NOT automatically start, but can be manually started

D. NOT automatically start and can NOT be manually started

Description of how HPCI is impacted by this loss of DC

HPCI can NOT be started since the power supply to several HPCI components has been lost, including power to the Aux Oil pump. Oil pressure supplied by the Aux Oil pump is required to open the HPCI governor valve. If the governor valve is not opened, the turbine will not turn. HPCI cannot be started in this condition. If HPCI had been running prior to the loss of 2R24-S022, it would continue to run.

A. **Incorrect**; see description above.
Plausible if the candidate does not know which components are supplied by this DC bus. If HPCI were running before the DC bus loss, the second part of this distractor would be a correct answer.

B. **Incorrect**; see description above.
Plausible if the candidate does not know which components are supplied by this DC bus. Guidance for local operation of HPCI is provided in 31RS-E41-001-2, "HPCI Operation From Outside The Control Room" If HPCI were already operating when the DC power was lost, HPCI operation would continue, however, local manual control would be the only way to control the system valves.

C. **Incorrect**; see description above.
Plausible if the candidate does not know which components are supplied by this DC bus or if the HPCI and RCIC systems are confused (RCIC can be manually started without DC power).

D. **Correct**; see description above.

SYSTEM: 206000 High Pressure Coolant Injection System

K6. Knowledge of the effect that a loss or malfunction of the following will have on the HIGH PRESSURE COOLANT INJECTION SYSTEM: (CFR: 41.7 / 45.7)

K6.02 D.C. power: BWR-2,3,4 . . . . . RO 3.3 / SRO 3.7*

Reference(s) used to develop this question:

31RS-E41-001-2, HPCI Operation From Outside the Control Room
34AB-R22-001-2, "Loss of DC Buses"
E41-HPCI-LP-00501, "HPCI Injection System" lesson plan
R42-ELECT-LP-02704 "DC Electrical"
EO 200.018.a.01

Reference(s) provided to the student:
None

8. 209001A4.04 001/2/1/CORE SPRAY CS/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 is at 100% power.

- o Both loops of Core Spray (CS) are in their normal standby lineup.
- o CS receives an automatic start signal

Which ONE of the following completes BOTH of these statements?

In their normal standby lineup, the CS Minimum Flow valves (2E21-F031A/B) are normally ___(1)___.

The CS Minimum Flow valves (2E21-F031A/B) will remain in their standby lineup until ___(2)___.

A.✓ (1) open

(2) discharge flow exceeds 950 gpm

B. (1) open

(2) the CS pump motor breaker is closed

C. (1) closed

(2) discharge flow exceeds 950 gpm

D. (1) closed

(2) the CS pump motor breaker is closed

Description: The CS minimum flow valves are normally in the OPEN position.
The min flow valves close when system flow goes above 950 gpm.

A. **Correct**; see description above

B. **Incorrect**; see description above
1st part is correct
2nd part is **plausible** since power available (pump breaker closed) to the pump motor is an input to the discharge valve logic. Additionally, the RHR minimum flow valves open when a low flow condition (<2310 gpm) exists as soon as the pump breaker is closed.

C. **Incorrect**; see description above
1st is **plausible** since HPCI and RCIC min flow valves are normally closed and the candidate may confuse which ECCS systems normally have open vs closed min flow valves.
2nd part is correct

D. **Incorrect,** see description above
1st is **plausible** since HPCI and RCIC min flow valves are normally closed and the candidate may confuse which ECCS systems normally have open vs closed min flow valves.
2nd part is plausible since power available (pump breaker closed) to the pump motor is an input to the discharge valve logic. Additionally, the RHR minimum flow valves open when a low flow condition (<2310 gpm) exists as soon as the pump breaker is closed.

SYSTEM: 209001 Low Pressure Core Spray System

A4 Ability to manually operate and/or monitor in the control room:
(CFR: 41.7 / 45.5 to 45.8)

A4.04 Minimal flow valves . . . . . RO 2.9/ SRO 2.9

Reference(s) used to develop this question:

E21-CS-LP-00801 Core Spray lesson plan
E11-RHR-LP-00701, "RHR System" lesson plan
34SV-SUV-018-1, ECCS Status Check
EO 008.002.a.05 State the interlocks associated with the following CS valves; a. Min Flow Valve F031

Reference(s) provided to the student:
None

9. 209001A4.05 001/2/1/CORE SPRAY CS/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 100% power, with HPCI tagged out for maintenance.

A trip of both reactor feedwater pumps results in a low reactor water level with conditions occurring as indicated in the following time line:

Time (minutes)

+0 min.	Reactor scram, all rods fully insert
+5 min.	Drywell pressure: 0.7 psig.
+5 min.	Reactor water level: -110 inches, decreasing at 0.5 inch/minute.
+5 min.	Reactor pressure: 960 psig.
+6 min.	Both Core Spray (CS) systems are in their normal standby lineups.

IAW 30AC-OPS-003-0, "Plant Operations", which ONE of the following describes the MINIMUM required operator actions (if any)?

A. manually start both CS pumps (only)

B. manually start both CS pumps and manually open discharge valves 2E21-F004A & F004B (only)

C. manually start both CS pumps and manually open discharge valves 2E21-F005A & F005B (only)

D. no manual operation of CS is required at this time, continue to monitor until system operation is required

Description: Core Spray pumps should have automatically started at -101 inches, but failed to do so. NMP-OS-007 requires the starting of ESF systems when they fail to auto start. 2E21-F004 is normally open in a standby lineup, so the action of manually opening the valve is not required. Interlocks prevent F005 from being manually opened at this pressure when 2E21-F004 valve is already opened. Starting the pump(s) is the MINIMUM action required.

A. **Correct**; see description

B. **Incorrect**; see description
1st part is correct.
2nd part is not correct since 2E21-F004 is normally open in a standby lineup, so the action of manually opening the valve is not required. **Plausibility** stems from the fact that the candidate must know which discharge valve is normally closed and that manual start section of the system operating procedure directs the operator to open F005 after starting the CS pump. If the candidate thinks the F004 valve is normally closed in standby, it is reasonable for them to think opening F004 is required (i.e. F004 confused with F005) by procedure.

C. **Incorrect** ; see description
1st part is correct, NMP-OS-007 requires the starting of ESF systems when they fail to auto start. 2nd part is not correct since interlocks prevent F005 from being manually opened at this pressure when 2E21-F004 valve is already opened. **Plausible** since the manual start of CS section of the system operating procedure directs the operator to open 2E21-F005 after starting the CS pump.

D. **Incorrect**; see description
NMP-OS-007 requires the starting of ESF systems when they fail to auto start. **Plausibility** is that the candidate may not have recognized the system has exceeded a start signal and failed to auto start, and in this case, differentiating between the procedure setpoint of -101 inches and the TS setpoint of -113 inches.

SYSTEM: 209001 Low Pressure Core Spray System

A4 Ability to manually operate and/or monitor in the control room:
(CFR: 41.7 / 45.5 to 45.8)

A4.05 Manual initiation controls . . . . RO 3.8 / SRO 3.6

Reference(s) used to develop this question:

34SO-E21-001-2 Core Spray System procedure
Unit 2 TS
E21-CS-LP-00801 Core Spray Lesson Plan
EO 008.002.a.01 From a list, select the actions necessary to manually start the CS system per NMP-OS-007, section 5.9.3.3

Reference(s) provided to the student:
None

10. 211000K4.03 001/2/1/SLC/NEW/FUND/HT2009-301/RO/ELJ/CME/

Concerning the **Unit 1** Standby Liquid Control (SLC) system, which ONE of the following completes both of these statements?

During normal operation, the primary means of ensuring that sodium pentaborate remains in solution inside the SLC tank is by the use of ___(1)___.

This ensures that the reactor will remain in at least ___(2)___ shutdown condition when SLC tank level has decreased to 32%, following SLC initiation during an Anticipated Transient Without a Scram (ATWS), regardless of control rod position.

A. (1) tank heaters

(2) Hot

B. (1) an air sparger

(2) Hot

C. (1) tank heaters

(2) Cold

D. (1) an air sparger

(2) Cold

Description: The SLC system uses tank heaters, boron concentration and water volume to maintain boron in solution. A bubbler level detector is used to determine level in the boron tank. An air sparger inside the boron tank is used to mix boron during boron addition and prior to taking samples. A caution tag is used to ensure the SLC pump is not operated when the air sparger is in use since it will cause pump cavitation.

35% SLC tank level corresponds to Hot Shutdown boron weight
14% SLC tank level corresponds to Cold Shutdown boron weight.

A. **Correct**; see description.

B. **Incorrect**; see description.
The 1st part is **plausible** since an air sparger inside the boron tank is used to mix boron during boron addition and prior to taking samples.
The 2nd part is correct.

C. **Incorrect**; see description
The 1st part is correct.
The 2nd part is **plausible** since the candidate must recall from memory the SLC tank level that corresponds to Hot vs. Cold Shutdown Boron Weight.

D. **Incorrect**; see description.
The 1st part is **plausible** since an air sparger inside the boron tank is used to mix boron during boron addition and prior to taking samples.
The 2nd part is plausible since the candidate must recall from memory the SLC tank level that corresponds to Hot vs. Cold Shutdown Boron Weight.

SYSTEM: 211000 Standby Liquid Control System

K4. Knowledge of STANDBY LIQUID CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K4.03 Keeping sodium pentaborate in solution . . . . . RO 3.8 / SRO 3.9

Reference(s) used to develop this question:

C41-SBLC-LP-01101 Standby Liquid Control lesson plan (EO 011.001.a.10)
EOP-CURVES-LP-2306, "EOP Curves and Limits"
EOP-CP3-LP-20327, "Level/Power Control (CP-3)" lesson plan (EO 201.092.a.02)
EOP-RCA-LP-20328, "RPV Control - ATWS (RCA) lesson plan (EOs 201.070.a.04 & 201.071.a.18)

Reference(s) provided to the student:
None

11. 212000A1.07 001/2/1/RPS CRD/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 2 was operating at 100% power

- o Control rod 26-35 is at notch position 48
- o The Rod Scram Switch for control rod 26-35, at 2H11-P610, is placed into the "DOWN (scram)" position
- o 2 seconds later, the switch is placed in the "UP (normal)" position.

Which ONE of the following is the expected Rod Position Indication at 2H11-P603 for control rod 26-35 after these actions are performed?

A. blank (over-travel)

B. 00

C. 24

D. 46

Description: Toggling this switch will de-energize the scram pilot solenoid valves for this rod only, and only when the switch is in the "down (scram)" position. Reactor pressure is adequate to cause rods to insert without a CRD pump running.

The rods will only move as long as the scram pilot solenoid valves are de-energized and 2 seconds is not long enough to fully insert control rods in this condition. When the switch is returned to the normal (up) position, the scram pilot solenoid valves will re-energize and the control rod will settle at the next notch position. There is not a seal in feature when individually scramming control rods. In this case, 2 seconds is not sufficient to move the control to a full in position.

Max. insertion time for a control rod is 7 seconds. 31EO-EOP-103-2 requires the switch be maintained in the "Up (scram)" position until rod movement stops to ensure the rod is fully inserted. Per TS, it takes up to 3.35 seconds to cause a rod that is at position48 to insert to position 06.

A. **Incorrect**; see description above.
Plausible since a scram does normally seal in and a scrammed rod window indication is initially blank until the scram is reset, or the SDV fills and CRDM piston dp equalizes.

B. **Incorrect**; see description above.
Plausible since 00 is the normal position the candidates see as rod position for a fully inserted control rod.

C. **Correct**; see description above.

D. **Incorrect**; see description above.
Plausible since this position corresponds to the rod position the rod would settle at if it were moved at normal drive speed.

SYSTEM: 212000 Reactor Protection System

A1. Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: (CFR: 41.5 / 45.5)

A1.07 Rod position information . . . . RO 3.4 / SRO 3.4

Reference(s) used to develop this question:

C71-RPS-LP-01001 RPS lesson plan
EO 010.020.a.02 Given a list, identify the statement that describes the effect on individual CRDMs of placing the individual control rod scram test switches in scram.
C11-CRDM-LP-00102, CRDM lesson plan
U2 TS table 3.1.4-1
31EO-EOP-103-2, "EOP Control Rod Insertion Methods"

Reference(s) provided to the student:
None

12. 215001A3.03 001/2/2/TIP/BANK MOD/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 is operating at 100% power with the "A" Traversing In-Core Probe (TIP) inserted in the core to perform 57CP-C51-010-0, "TIP Flux Probing Monitor".

A transient occurs on **Unit 1** with the following plant conditions:

Reactor pressure ..... 900 psig and stable
Reactor level (lowest) ..... -20 inches and slowly increasing
Drywell pressure ..... 1.5 psig and stable
Drywell temperature ..... 129°F

Which ONE of the following completes the statement below?

With NO operator action, the "A" TIP will _____ :

A. remain in the core with its Ball valve OPEN and its Shear valve ACTUATED.

B. remain in the core with its Ball valve OPEN and its Shear valve NOT ACTUATED..

C. withdraw to the "in-shield" position with its Ball valve CLOSED and its Shear valve ACTUATED..

D. withdraw to the "in-shield" position with its Ball valve CLOSED and its Shear valve NOT ACTUATED.

Description: The TIP receives a signal to withdraw to the "in-shield" position upon receipt of a group 2 signal (1.85 psig DW press or +3" RPV water level. The ball valve auto closes when the probe is fully withdrawn.

A. **Incorrect**; see description above
Plausible since the TIP shear valve is actuated if the TIP were stuck in the core with evidence of a leak via that pathway..

B. **Incorrect**; see description above
Plausible since the candidate must know that the withdrawal occurs at + 3" rather than -35", -60" or -101" inches

C. **Incorrect**; see description above
Plausible since the TIP shear valve is actuated if the TIP were stuck in the core with evidence of a leak via that pathway..

D. **Correct**; see description above

SYSTEM: 215001 Traversing In-Core Probe

A3. Ability to monitor automatic operations of the TRAVERSING IN-CORE PROBE including: (CFR: 41.7 / 45.7)

A3.03 Valve operation: Not-BWR1 . . . . . RO 2.5* / SRO 2.6*

Reference(s) used to develop this question:

T23-PC-LP-01301 Primary Containment lesson plan
EO (013.036.A.02, 013.039.a.04)
EO (013.007.a.03, 013.012.a.02, 013.019.a.02, 013.020.a.02)

Reference(s) provided to the student:
None

13. 215002K6.04 001/2/2/RBM/BANK/FUND/HT2009-301/RO/ELJ/CME/

Unit 2 APRM channel "B" was bypassed with the reactor at rated conditions.

What effect will this action have on the RBM system?

- The "B" RBM _____

A. will be INOP due to a loss of its reference APRM.

B. reference APRM will automatically shift to APRM channel D

C. reference APRM will automatically shift to APRM channel C

D. reference APRM will automatically shift to APRM channel A

Description: When the RBM loses its primary reference APRM it automatically selects the first alternate APRM. D APRM is the first alternate, second alternate is C APRM.

A. **Incorrect**
The RBM automatically selects the first alternate APRM

B. **Correct**
D APRM is the first alternate, second alternate is C APRM.

C. **Incorrect**
C APRM is the second alternate, not the first alternate

D. **Incorrect**
A APRM is only used as a reference for the A RBM channel

SYSTEM: 215002 Rod Block Monitor System

K6. Knowledge of the effect that a loss or malfunction of the following will have on the ROD BLOCK MONITOR SYSTEM : (CFR: 41.7 / 45.7)

K6.04 APRM reference channel: BWR-3,4,5 . . .. RO 2.8 / SRO 3.0

Reference(s) used to develop this question:

C51-PRNM-LP-01203, "Power Range Neutron Monitoring System"

Reference(s) provided to the student:
None

14. 215003A1.03 001/2/1/IRM RPS/BANK MOD/HIGHER/HT2009-301/RO/ELJ/CME/

A **Unit 1** Reactor Startup is in progress.

IRMs read as follows:

IRM	Reading	Range	IRM	Reading	Range
A	38/125	8	**B**	35/125	8

Which ONE of the following predicts how the plant will respond to the following switch manipulations?

o IRM "A" range switch is placed in the RANGE 7 position.
o IRM "B" range switch is placed in the RANGE 7 position.

The IRMs will _____.

A. NOT initiate any automatic action

B. initiate a rod block (only)

C. initiate a rod block and a half scram (only)

D. initiate a full scram

Description: The candidate must know or correctly calculate the following setpoints to correctly answer the question:
Rod block on the odd numbered range switches: 25.6/40
Rod block on the even numbered range switches: 80/125
Reactor Scram signal on the odd numbered range switches: 36.8/40
Reactor Scram signal on the even numbered range switches: 115/125

The candidate must know that the reading on an odd numbered range is numerically the same as the next higher range.
In this question, the "A" channel is above the scram setpoint on range 7 (reading 38/40) and the "B" channel is above the rod block setpoint, but below the scram setpoint on range 7 (reading 35/40)

Rod block setpoint is added to provide plausibility for the 3rd distractor

A. **Incorrect**; see description above
Plausibility is based on the candidates knowledge of how the indications and setpoints are impacted by changing ranges on the IRMs. The candidate must perform the correct calculation of the setpoints candidates often have trouble transitioning from a 0 to 40 scale from a 0 to 125 scale).

B. **Incorrect**; see description above
Plausibility is based on the candidates knowledge of how the indications and setpoints are impacted by changing ranges on the IRMs. The candidate must perform the correct calculation of the setpoints candidates often have trouble transitioning from a 0 to 40 scale from a 0 to 125 scale).

C. **Correct**; see description above

D. **Incorrect**; see description above
Plausibility is based on the candidate knowledge of how the indications and setpoints are impacted by changing ranges on the IRMs. The candidate must perform the correct calculation of the setpoints (i.e. 40 on a 125 scale rather than 40 on a 100 scale which would indicate a scram setpoint of 32/40 rather than 36.8/40) (Note, a rod block would be in if a full scram did occur (SDV level), but not be due to the IRMs).

SYSTEM: 215003 Intermediate Range Monitor (IRM) System

A1. Ability to predict and/or monitor changes in parameters associated with operating the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM controls including:
(CFR: 41.5 / 45.5)

A1.03 RPS status . . . . . RO 3.6 / SRO 3.7

Reference(s) used to develop this question:

34AR-603-221-1 IRM Bus A Upscale Trip or Inop ARP
34AR-603-221-1 IRM Upscale ARP
C51-IRM-LP-01202, Intermediate range Monitors
EO (012.003.C.09, 012.007.A.02)
EO (012.003.C.010, 012.007.B.02)

Reference(s) provided to the student:
None

15. 215004K6.01 001/2/1/SRM RPS/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 is starting up.

- All Intermediate Range Monitors (IRM) are on RANGE 4 (reading on-scale)
- All Source Range Monitors (SRM) are all partially withdrawn and indicating 5×10^4 counts per second (cps)
- The "1A" SRM High Voltage power supply experiences a failure that causes its voltage decrease from 350 VDC to 20 VDC, resulting in the "1A" SRM count rate decreasing to 4 cps

A Control Rod Block ___(1)___ occur because ___(2)___.

A. (1) will

(2) the "1A" SRM is partially withdrawn

B. (1) will

(2) of an INOP trip on the "1A" SRM

C. (1) will NOT

(2) of the IRM range switch positions

D. (1) will NOT

(2) the "1A" SRM is reading 4 cps

Description: SRM high voltage low will result in an Inop rod withdraw block.
This rod block is bypassed when IRMs are on range 8 or higher.
The partially withdrawn SRM detector rod block is bypassed because the IRM range switches positioned on a range are above range 3.
The SRM downscale is bypassed when the IRM range switches positioned on a range are above Range 3.
TS operability requirements is 3 cps for the downscale rod block setpoint.

A. **Incorrect**; see description above
1st part is correct.
2nd part is **plausible** since a partially withdrawn SRM detector will result in a rodblock if the IRM range switches are on range 1 or 2.

B. C**orrect**; see description above

C. **Incorrect**; see description above
Both parts are incorrect.
Plausible since the candidate may focus on the 4 cps (downscale) SRM indication (overlook the Inop due to the high voltage low condition). The downscale rod block is bypassed since the range switches are on range 4.

D. **Incorrect**; see description above
Both parts are incorrect.
Plausible since the indication is above the required TS limit of 3 cps for the downscale rod block.

SYSTEM: 215004 Source Range Monitor (SRM) System

K6. Knowledge of the effect that a loss or malfunction of the following will have on the SOURCE RANGE MONITOR (SRH) SYSTEM : (CFR: 41.7 / 45.7)

K6.05 Trip Units . . . . . RO 2.6 / SRO 2.8

Reference(s) used to develop this question:

C51-SRM-LP-01201 SRM lesson plan
Unit 1 TS Surveillance Requirement SR 3.3.1.2.4

Reference(s) provided to the student:
None

16. 215005G2.1.20 001/2/1/OPRM/NEW/HIGHER/HT 2009-301/RO/ELJ/CME/

Unit 1 is operating at 40% power in Single Recirc Loop Operation (SLO).

- o The "1B" Reactor Recirculation Pump is in service
- o The team is performing actions to re-start the "1A" Reactor Recirculation pump
- o Both Circulating Water pumps are in service
- o All Oscillation Power Range Monitors (OPRMs) are inoperable due to a calibration error
- o The breaker supplying power to "1B" 4160 VAC bus trips.

Which ONE of the following describes the required procedural actions?

A. Manually scram the reactor.

B. Maintain current power level and re-start the "1A" Recirculation Pump.

C. Reduce reactor power by inserting control rods in reverse order of the Pull Sheets to stabilize Main Condenser Vacuum >25" Hg Vac.

D. Reduce reactor power by inserting control rods in reverse order of the Pull Sheets as required to remain in the allowable region of the Power to Flow map.

Description: When 4160 "B" supply breaker trips, the only operating Reactor Recirc pump trips. A condition which requires a manual scram then exists because all OPRMs are inoperable. A shutdown is not immediately required when OPRMs are operable (allowed to operate for 12 hrs per TS). One Circ Water pump continues to operate and will be adequate to maintain vacuum for this power level.

A. **Correct;** see description above.

B. **Incorrect;** see description above.
Plausible since restarting a Recirc Pump is a correct action if the OPRMs were operable and a scram is not inserted.

C. **Incorrect;** see description above.
Plausible if the candidate does not know of the requirement to scram with the OPRMs inop without forced circulation and assumes that one Circ Water pump is not adequate to maintain condenser vacuum at this power level.

D. **Incorrect;** see description above.
Plausible since this is a required action if the unit were not scrammed (i.e. the OPRMs were operable).

SYSTEM: 215005 Average Power Range Monitor/Local Power Range Monitor System

2.1 Conduct of Operations

2.1.20 Ability to interpret and execute procedure steps. (CFR: 41.10 / 43.5 / 45.12) | IMPORTANCE RO 4.6 / SRO 4.6

Reference(s) used to develop this question:

B31-RRS-LP-00401, "Reactor Recirculation System"
EO 200.037.a.02
34AB-B31-001-1 "Trip of One or Both Reactor Recirculation Pumps"
34AB-R22-004-1 "Loss of 4160V Bus 1A, 1B, 1C or 1D"

Reference(s) provided to the student:
None

17. 215005G2.4.11 001/2/1/APRM EOP/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 is operating at 40% power when a scram signal is received.

- o Some control rods do NOT fully insert
- o APRMs are all indicating 3% (steady)
- o Both Reactor Recirculation pumps are running at minimum speed
- o No entry conditions for the EOP flowcharts have been met or exceeded

IAW 34AB-C11-005-1, "Control Rod Insertion Methods", which ONE of the following completes both of these statements?

- o Overriding of all automatic scram signals to insert control rods is _____(1)_____.
- o Procedurally, tripping of the Reactor Recirculation pumps is _____(2)_____.

A. (1) allowed

(2) required

B. (1) allowed

(2) NOT required

C. (1) NOT allowed

(2) required

D. (1) NOT allowed

(2) NOT required

Description: Without EOP entry conditions being exceeded, with power less than 5%, a scram is controlled by use of abnormal procedures, even if all rods do not insert. If power were above 5%, 31EO-EOP-103-1 would be used to insert control rods. In this case 34AB-C71-001-1 directs operators to insert rods per 34AB-C11-005-1. This procedure does NOT allow overriding of "all" automatic scram signals (the SDV high level trip is allowed to be overridden). The scram procedure (34AB-C71-001-1) directs Recirc Pumps to be started if they are tripped.

A. **Incorrect**
1st part is not correct; this action is not allowed when using 34AB-C11-005-1 to insert control rods. **Plausible** since this is a required action for repeating a manual scram when using 31EO-EOP-103-1 to insert control rods.
2nd part is not correct, 34AB-C71-001-1 actually requires the Recirc Pumps to be restarted if tripped, this will minimize thermal stratification within the RPV. **Plausible** since the Recirc pumps would be manually tripped if RCA were in use with power above 5% in order to reduce reactor power.

B. **Incorrect**
1st part is not correct; this action is not allowed when using 34AB-C11-005-1 to insert control rods. **Plausible** since this is a required action for repeating a manual scram when using 31EO-EOP-103-1 to insert control rods.
2nd part is correct.

C. **Incorrect**
1st part is correct
2nd part is not correct, 34AB-C71-001-1 actually requires the Recirc Pumps to be restarted if tripped, this will minimize thermal stratification within the RPV. **Plausible** since the Recirc pumps would be manually tripped if RCA were in use with power above 5% in order to reduce reactor power.

D. **Correct**

SYSTEM: 215005 Average Power Range Monitor/Local Power Range Monitor System

2.4 Emergency Procedures / Plan

2.4.11 Knowledge of abnormal condition procedures. (CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 4.0 SRO 4.2

Reference(s) used to develop this question:

34AB-C11-005-1, Control Rod Insertion Methods
34AB-C71-001-1, Scram Procedure

Reference(s) provided to the student:
None

18. 217000K2.03 001/2/1/RCIC/NEW/FUND/HT2009-301/RO/ELJ/CME/

Which ONE of the following is the power supply to the **Unit 2** RCIC flow controller?

A. Vital AC (2R25-S063)

B. DC Cabinet "2A" (2R25-S001)

C. Instrument Bus "2A" (2R25-S064)

D. Reactor Bldg ESS MCC "2B" (2R24-S022)

Description: The RCIC controller is supplied AC power via an inverter from 125V DC Cabinet "2A".

A. **Incorrect;** see description above
Plausible since Vital AC supplies power so many Control Room instruments. Battery provides backup.

B. **Correct,** see description above

C. **Incorrect,** see description above
Plausible since Instrument Bus "2A" supplies logic power for RCIC at the RSDP.

D. **Incorrect,** see description above
Plausible since this bus supplies HPCI components (including the HPCI flow controller via 2R24-S002) and can be confused by the candidate. If this were changed to Reactor Bldg ESS MCC "2A" (2R24-S021) it would be a correct answer.

SYSTEM: 217000 Reactor Core Isolation Cooling System (RCIC)

K2. Knowledge of electrical power supplies to the following: (CFR: 41.7)

K2.03 RCIC flow controller . . . . . RO 2.7* / SRO 2.8*

Reference(s) used to develop this question:

34AR-602-319-2 RCIC Inverter K603 Power Failure
34AB-R25-002-2 Loss of Instrument Buses
34AB-R25-001-2 Loss of Vital AC Bus
E41-HPCI-LP-00501 HPCI lesson plan

Reference(s) provided to the student:
None

19. 218000K1.06 001/2/1/ADS/BANK MOD/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 has experienced a LOSP.

The following conditions exist:

- o Reactor..... All rods in
- o All 4160 VAC buses..... De-energized
- o RPV Pressure..... 700 psig
- o RWL..... -135 inches
- o Drywell Pressure..... 3 psig
- o ADS Inhibit Switches..... "Normal" position
- o These conditions are maintained for 5 minutes

Immediately after the 5 minutes has elapsed, an operator starts the "2A" Emergency Diesel.

- "2A" Core Spray (CS) pump starts.
- "2A" CS develops normal discharge pressure for shut off head operation.

Which ONE of the following identifies when all the ADS valves will automatically open?

A.✓ Immediately

B. 102.5 seconds later

C. 11 minutes later

D. 12 minutes and 42.5 seconds later

Refer to logic drawing provided as a reference for developing this test item.
Description: The 102.5 second timer began timing as soon as the following conditions existed concurrently: D/W pressure >1.85 psig and RPV level <-101". As a result, with the exception of low pressure ECCS confirmatory pressure, all other necessary logic connections are made to auto open the ADS valves without a time delay.

A. **Correct**; see description above

B. **Incorrect**; see description above
Plausible if the candidate assumes the CS permissive is part of the 102.5 sec timer. The low pressure ECCS pressure permissive must be present after the 102.5 sec timer times out in order for the ADS valves to auto open.

C. **Incorrect**; see description above
The 11 minute timer initiated and sealed in as soon as RPV level dropped below -101". CS is not part of the 11 minute timer initiation logic. **Plausible** if the candidate assumes the CS permissive is part of the 11 min. timer.
The low pressure ECCS pressure permissive must be present after the 11 minute and 102.5 sec timers time out in order for the ADS valves to auto open.

D. **Incorrect**; see description above
The 102.5 second timer began timing as soon as the following conditions existed concurrently: D/W pressure >1.85 psig and RPV level <-101". The 11 minute timer initiated and sealed in as soon as RPV level dropped below -101".
Plausible if the candidate assumes that all conditions must be met for the timers begin to time out or if the candidate assumes the CS permissive is part of the 11 min. timer. The low pressure ECCS pressure permissive must be present after the 102.5 sec timer times out in order for the ADS valves to auto open.

SYSTEM: 218000 Automatic Depressurization System

K1. Knowledge of the physical connections and/or cause/effect relationships between AUTOMATIC DEPRESSURIZATION SYSTEM and the following:
(CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.06 Safety/relief valves . . . . . RO 3.9* / SRO 3.9*

Reference(s) used to develop this question:

B21-ADS-LP-0381 "Auto Depress System" EO 038.004.a.02
B21-ADS-03801 Fig 2 & Fig 4
34SO-B21-001-2, "ADS and LLS System" procedure

Reference(s) provided to the student:
None

20. 223001K3.09 001/2/2/RWL INDICATION/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 100% power.

- o A complete loss of Drywell (D/W) Cooling results in D/W temperature increasing from 130°F to 285°F

If all other parameters remain the same, which ONE of the following describes the relationship between indicated and actual reactor water level (RWL) for the 2B21-R604 and 2C32-R606 instruments when monitoring RWL at the 2H11-P603 panel?

	2B21-R604A and B	2C32-R606A, B and C
A.	indicated = actual	indicated > actual
B.*	indicated > actual	indicated > actual
C.	indicated < actual	indicated < actual
D.	indicated = actual	indicated < actual

Description: DW temp increasing will result in a decreasing density in the reference leg. As reference leg density decreases due to heating, with the variable leg remaining relatively constant, dp will decrease. Decreasing dp will result in a higher indicated level, even with no change in actual level.

Plausibility: Candidates historically have difficulty remembering (i.e. common misconception) the relationship between heating reference legs, dp and indicated level vs actual level.

A. **Incorrect**; see description above.
1st part is not correct. **Plausible** since these reference legs are heated.
2nd part is correct.

B. **Correct**; see description above

C. **Incorrect**; see description above
Both parts are incorrect.
Both parts are **plausible** since candidates historically have difficulty remembering (i.e. common misconception) the relationship between heating reference legs, dp and indicated level vs actual level.

D. **Incorrect**; see description above
Both parts are incorrect.
1st part is plausible since these reference legs are heated.
2nd part is **plausible** since candidates historically have difficulty remembering (i.e. common misconception) the relationship between heating reference legs, dp and indicated level vs actual level.

SYSTEM: 223001 Primary Containment System and Auxiliaries

K3. Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT SYSTEM AND AUXILIARIES will have on following: (CFR: 41.7 / 45.4)

K3.09 Nuclear boiler instrumentation . . . . RO 2.8 / SRO 3.1

Reference(s) used to develop this question:

B11-RXINS-LP-04404, "Reactor Vessel Instrumentation"
EO 200.002.a.014

Reference(s) provided to the student:
None

21. 223002K3.11 001/2/1/HPCI/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is at 100% power with the Main Control Room Environmental Control (MCREC) System aligned in the NORMAL mode.

The following annunciators are received:

- o **MAIN STEAM LINE FLOW A HIGH (603-214-2)**
- o **MAIN STEAM LINE FLOW B HIGH (603-215-2)**
- o **GROUP I SYSTEM A TRIP (603-208-2)**
- o **GROUP I SYSTEM B TRIP (603-209-2)**

IAW 34SO-Z41-001-1, "Control Room Ventilation System" the MCREC System _____.

A. must be aligned to the Purge mode

B. must be aligned to the Isolation mode

C. may remain aligned in the Normal mode

D. must be aligned to the Pressurization mode

Description; High Main Steam flow (of 169 psid) is one of the setpoints used to automatically shift the MCREC system into Pressurization mode.

A. **Incorrect**; see description above.
Plausible since purge mode is a valid mode of MCREC and the candidate must know when this mode of operation is required to be used.

B. **Incorrect**; see description above.
Plausible since isolation mode is a valid mode of MCREC and the candidate must know when this mode of operation is required to be used.

C. **Incorrect**; see description above.
Plausible since the candidate must know that high steam flow setpoint for pressurization mode is the same as the setpoint for the high steam flow annunciators to eliminate this distractor.

D. **Correct**; see description above.

SYSTEM: 223002 Primary Containment Isolation System/Nuclear Steam Supply Shut-Off

K3. Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: (CFR: 41.7 / 45.4)

K3.11 Plant ventilation . . . . . RO 2.8 / SRO 2.9

Reference(s) used to develop this question:

T41-SCHVAC-LP-01303, SECONDARY CONTAINMENT VENTILATION, 037.008.a.02
34SO-Z41-001-1, "Control Room Ventilation System"
34AR-603-214/215-2, "MAIN STEAM LINE FLOW A/B HIGH" ARPs

Reference(s) provided to the student:
None

22. 226001K2.02 001/2/2/RHR E11/BANK/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 was operating at 100% power.

- o The Alternate Supply Breaker to 4160VAC bus "1E" is tagged out.

A loss of Startup Transformer (SAT) "1D" occurred.

- o Torus Pressure reaches 3 psig during the transient.
- o "1A" and "1B" RHR pumps running, both being used to spray the Torus.

The power supply for the 4160 VAC bus to the:

- "1A" RHR Pump is __(1)__.
- "1B" RHR Pump is __(2)__.

A. (1) SAT "1C"

(2) SAT "1C"

B. (1) its associated EDG

(2) SAT "1C"

C. (1) SAT "1C"

(2) its associated EDG

D. (1) its associated EDG

(2) its associated EDG

Description: Startup auxiliary transformer (SAT) "1D" is the normal supply to all U1 emergency buses. SAT "1C is the alternate supply to the emergency buses. EDGs are the emergency supply in case there is a loss of both normal and alternate supplies. Ordinarily the emergency buses automatically transfer to alternate, however in this case the "1E" (power to the "1A" RHR pump) bus alternate supply breaker is tagged out, so when the "1D" SAT is lost, the "1A" EDG will start and supply the "1E" emergency bus. The "1G" emergency bus (power supply to the "1B" RHR pump) will be supplied power via the alternate supply breaker (SAT "1C").

A. **Incorrect**; see description above.
1st part is not correct. **Plausible** if the candidate thinks that SAT "1C" is the normal supply to the emergency buses.
2nd part is correct.

B. **Correct**; see description above.
1st part is correct.
2nd part is correct

C. **Incorrect**; see description above.
1st part is not correct; **plausible** since the examinee must know which bus supplies power to the pump to eliminate this distractor.
2nd part is not correct; **plausible** since the examinee must know which bus supplies power to the pump to eliminate this distractor.

D. **Incorrect**; see description above.
1st part is correct.
2nd part is not correct; **plausible** since all U1 EDGs start when SAT "1D" is lost. The candidate may assume they start and tie to the emergency buses.

KA:

SYSTEM: 226001 RHR/LPCI: Containment Spray System Mode

K2. Knowledge of electrical power supplies to the following: (CFR: 41.7)

K2.02 Pumps . . . . . RO 2.9* / SRO 2.9*

Reference(s) used to develop this question:

R22-4160 VAC-LP-02702 4160 VAC lesson plan, EO 200.017.aa.01
34SO-R22-001-1, "4160 VAC System" procedure

Reference(s) provided to the student:
None

23. 233000G2.2.25 001/2/2/FPC/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 Fuel Pool level is required to be lowered, while Fuel movement is in progress.

IAW with Tech Spec Limiting Condition for Operation (LCO) 3.7.8, Spent Fuel Storage Pool Water Level, which ONE of the following is the LOWEST level that will still meet LCO 3.7.8 WITHOUT entering a Required Action Statement (RAS), when lowering the fuel pool water level?

A. 23.1 feet

B. 22.1 feet

C. 21.1 feet

D. 20.1 feet

Description: During movement of irradiated fuel in the spent fuel pools, U1 TS requirement for fuel pool level is >= 21ft over top of irradiated assemblies seated in the spent fuel storage racks.

A. **Incorrect**; see description above.
Plausible since 23 feet is the limit for minimum water level maintained above irradiated fuel seated within the RPV (TS 3.9.6).

B. **Incorrect**; see description above.
Plausible since RHR operability requirements change above/below 22ft 1/8 inch (TS 3.9.7).

C. **Correct**; see description above.
(TS 3.7.8)

D. **Incorrect**; see description above.
Plausible since this is the lowest level that does require a RAS.
(TS 3.7.8)

SYSTEM: 233000 Fuel Pool Cooling and Clean-up

2.2 Equipment Control

2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2) IMPORTANCE RO 3.2 / SRO 4.2

Reference(s) used to develop this question:

TS section 3.7.8
TS section 3.9.6
TS section 3.9.7

Reference(s) provided to the student:
None

24. 239001K5.09 001/2/2/DECAY HEAT/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 2 was at 100% power when a reactor scram and a Main Turbine trip occurred.

The operators complete all steps on the following placards:

- RC-1
- RC-2
- RC-3
- TC-1

SELECT the component that will be contributing to RPV decay heat removal?

A. Reactor feed pump low pressure steam supply.

B. Reactor feed pump high pressure steam supply.

C. Moisture Separator Reheater first stage steam supply.

D. Moisture Separator Reheater second stage steam supply.

Description: The RFP low pressure steam is isolated when the main turbine is tripped. TC-1 isolates the MSR 2nd stage by closing the RSSVs. The MSR 1st stage steam supply is isolated by the turbine stop valve closure, which is completed per TC-1. The RFP high pressure steam supply (main steam) is still available.

A. **Incorrect,** see description above.
(LP supply) **Plausible** since the RFPT does have two sources of steam. The low pressure supply is designed to be the primary steam source to the RFP at rated power. The LP control valve fully opens before the HP control valve opens.

B. **Correct;** see description above.

C. **Incorrect,** see description above.
(1st stage MSRs) **Plausible** since the RSLLVs are still be open and had the operator not remembered that the RSSVs were on TC-1 then this area of the MSRs would still be using steam.

D. **Incorrect,** see description above.
(2nd stage MSRs) **Plausible** because it does use main steam from the high pressure turbine. Had the main turbine not been tripped manually or automatically it would be using steam. It also requires the candidate to remember the source of MSR first stage heating steam.

SYSTEM: 239001 Main and Reheat Steam System

K5. Knowledge of the operational implications of the following concepts as they apply to MAIN AND REHEAT STEAM SYSTEM: (CFR: 41.5 / 45.3)

K5.09 Decay heat removal . . . . . RO 3.4 / SRO 3.5

Reference(s) used to develop this question:

34AB-C71-001-2, "Scram Procedure"
N21-CNDFW-LP-00201, "Condensate and Feedwater System" lesson plan

Reference(s) provided to the student:
None

25. 239002A3.01 001/2/1/ADS LIGHTS/BANK MOD/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 has experienced a transient and the Automatic Depressurization System (ADS) automatically actuated.

- o Reactor pressure decreases from 1000 psig to 50 psig.
- o No ADS switches have been manipulated.

Which ONE of the following describes ALL of the lights that are illuminated for each ADS valve?

The Red _____ illuminated.

A. light ONLY is

B. and Green lights are

C. and Amber lights are

D. , Amber and Green lights are

Description: When ADS actuates, the associated red light illuminates indicating power is sent to the solenoid valve. The red light remains energized until operator action is taken to reset the ADS logic.

The SRV Amber light is actuated by a SRV tail pipe pressure switch, and this light is sealed in until reset by a switch on the 1H11-P602 panel.

The green light is automatically extinguished by the ADS logic when ADS actuates.

Plausibility; candidates historically confuse ADS light indications vs LLS light indications. The amber light will not illuminate if reactor pressure decreases to below the mid 200s pressure range when an SRV is opened..

A. **Incorrect,** see description above.
Plausible if the candidate assumes the amber light automatically extinguishes when tailpipe pressure is decreases to below 85 psig. This would be a correct answer if ADS had initiated with low reactor pressure (below mid 200 psig range) since the amber light does not illuminate at RPV pressures below this pressure range.

B. **Incorrect,** see description above.
Plausible if the candidate confuses the difference between ADS and LLS indications and assumes the amber light automatically extinguishes when tailpipe pressure decreases to below 85 psig. The green light remains illuminated when LLS actuates.

C. **Correct,** see description above.

D. **Incorrect,** see description above.
Plausible if the candidate confuses the difference between ADS and LLS indications for ADS and sensor actuated indications, but remembers that the amber light seals in. The green light remains illuminated when LLS actuates.

SYSTEM: 239002 Relief/Safety Valves

A3. Ability to monitor automatic operations of the RELIEF/SAFETY VALVES including:
(CFR: 41.7 / 45.7)

A3.01 SRV operation after ADS actuation . . . . . RO 3.8* / SRO 3.9*

Reference(s) used to develop this question:

B21-ADS-LP-03801, "Automatic Depressurization System" lesson plan
EO 038-002.a.02
34SO-B21-001-1, "ADS and LLS Systems" procedure

Reference(s) provided to the student:
None

26. 241000K1.25 001/2/2/TURB RUNBACK/BANK MOD/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 is starting up, currently at 35% power.

The Main Generator system parameters are listed below:

- o Power ..... 300 MWe
- o Voltage ..... 24,000 volts
- o Current ..... 7,000 amps

Stator Cooling system parameters associated with the Main Generator are listed below:

- o Generator Inlet Flow ..... 500 gpm
- o Generator Inlet Pressure ..... 34 psig
- o Generator Outlet Temperature .... 84°C

These conditions will result in the _____

A. bypass valves remaining closed with the generator remaining on line.

B. bypass valves being throttled partially open with the generator remaining on line.

C. turbine control valves throttling closed, followed by a high neutron flux Reactor Scram.

D. turbine control valves throttling closed, followed by a high reactor pressure Reactor Scram.

Description; The runback will cause the Main Turbine control valves to close to 24% main generator load. The impact on the plant is determined by whether the sufficient pressure control margin exists to maintain pressure, rather than resulting in a significant pressure increase. If power is above the capacity of the system to maintain pressure (approx 50%), pressure will increase and the plant will either scram due to high pressure psig or high neutron flux.
The limits for this question include the runback setpoint of 24% load, 21% bypass valve capacity(BPV), and 5%steam loads for RFPTs, SJAE, Gland seal condenser realizing this value will be between 2-5% at less than 100% power. Therefore; 35% power is within the capacity of the pressure control system following the runback.

A. Incorrect, see description above.
Plausible if the candidate assumes the turbine control valves and existing steam loads can maintain system pressure without the assistance of the Bypass Valves.

B. Correct, see description above.

C. incorrect, see description above.
Plausible since pressure does increase as a result of the runback.

D. Incorrect, see description above.
Plausible since neutron flux does increase as pressure increases.

SYSTEM: 241000 Reactor/Turbine Pressure Regulating System

K1. Knowledge of the physical connections and/or cause/effect relationships between REACTOR/TURBINE PRESSURE REGULATING SYSTEM and the following:
(CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.25 Stator water cooling: Plant-Specific . . . . . RO 2.8 / SRO 2.8

Reference(s) used to develop this question:

B21-SLLS-LP-01401,
N43-SWC-LP-02301, "Stator Water Cooling" EO 200.089.a.01
N32 DEHC-LP-01902
34SO-N43-003-1, "Stator Water Cooling System"

Reference(s) provided to the student:
None

27. 259002A4.01 001/2/1/RFPT/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is in Mode 2, performing a Startup.

- o Reactor pressure 900psig
- o Reactor power 3%
- o Reactor Water Level (RWL) ..... 37 inches
- o RWL control ..... Reactor Feed Pump Turbine (RFPT) "2A", in automatic at 2800 rpm.
- o RFPT "2B" ..... Tagged out of service

Due to intermittent problems with the "2A" RFPT M/A station, 2C32-R601A, Maintenance has requested permission to take 2C32-R601A to the maintenance shop for repair.

IAW 34SO-N21-007-2, "Condensate and Feedwater System", which ONE of the following describes the affect on plant startup if the M/A Station is removed for repair?

A. Plant startup CANNOT continue. Reactor pressure must be reduced to 400 psig.

B. Plant startup CANNOT continue. RFPT "2A" speed must be reduced below 2100 rpm.

C. Plant startup CAN continue up to the point where RFPT "2A" flow reaches 7.0 Mlbm/hr.

D. Plant startup CAN continue, up to the point where RFPT "2A" reaches 6300 rpm

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Description: Operation of the RFP on the Speed Setter will allow RFP operation up to the high speed stop of 5800 RPM. Overspeed trip occurs at a point between 5900 - 6300 RPM. 34SO-N21-007 contains a section, "RFPT Alternate Startup" which allows the startup to continue with RFP operation controlled by the speed setter.

A. Incorrect, see description above.
Plausible since 400 psig will restore RPV pressure to below the discharge head of a Condensate Booster pump.

B. Incorrect, see description above.
Plausible since 2100 rpm is the speed referenced within the Condensate and Feedwater procedure for transferring control of the RFP from the speed setter to the M/A station.

C. Correct, see description above.

D. Incorrect, see description above.
Plausible if the candidate assumes the speed may be increased to the overspeed trip setpoint using the speed setter controls. This is the switch used to set the overspeed trip setpoint.

SYSTEM: 259002 Reactor Water Level Control System

A4. Ability to manually operate and/or monitor in the control room:
(CFR: 41.7 / 45.5 to 45.8)

A4.01 All individual component controllers in the manual mode . . . . . RO 3.8 / SRO 3.6

Reference(s) used to develop this question:

34SO-N21-007-2, Attachment 8 refers to the high speed stop of 5800 rpm.
N21-CNDFW-LP-00201, Condensate and Feedwater System" EO 002.027.a.01

Reference(s) provided to the student:
None

28. 261000A2.12 001/2/1/SBGT RAD/BANK MOD/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is in a Refueling Outage with fuel movement in progress.
Minor damage to a fuel bundle results in an increase in Refueling Floor radiation levels.

The "Refueling Floor Vent Exhaust" monitors are reading:

- o 2D11-K634A - 19 mRem/hr
- o 2D11-K634B - 18 mRem/hr
- o 2D11-K634C - 15 mRem/hr
- o 2D11-K634D - 14 mRem/hr

Which ONE of the following predicts BOTH the impact of these conditions on the Unit 1 and Unit 2 Standby Gas Treatment Systems (SBGT) <u>and</u> any procedural actions required for mitigation of the event.

The total number (Unit 1 and Unit 2 combined) of SBGTs that will start is ___(1)___.

___(2)___ ventilation systems will be verified tripped and isolated.

A. (1) two (ONLY)

(2) ONLY the Unit 1 and Unit 2 Refueling Floor

B. (1) four

(2) ONLY the Unit 1 and Unit 2 Refueling Floor

C. (1) two (ONLY)

(2) BOTH the Unit 1 and Unit 2 Refueling Floor and Reactor Building

D. (1) four

(2) BOTH the Unit 1 and Unit 2 Refueling Floor and Reactor Building

Description: There are 12 Refuel Floor Process Radiation monitors.

Setpoints:
- o 2D11-K634A-D rad monitors for U2 .... 5.7 mR/hr
- o 2D11-K611A-D rad monitors for U2 .... 18 mR/hr
- o 2D11-K635A-D rad monitors for U2 .... 6.9 mR/hr

With all 4 of the 2D11-K634 rad monitors above the setpoint, all 4 SBGT systems will auto start.

For **plausibility**, the following system operation was considered. Exceeding the setpoint on the "A" and "B" monitors auto start only SBGT "A." The system requires monitors "C" and "D" to reach their setpoints to auto start the "B" fan of SBGT. U1 only has 4 process rad monitors for the Refuel Floor with a setpoint of 18 mR/hr

A. Incorrect, see description above.
Plausible since the candidate may confuse the rad monitors and their setpoints. If the candidate remembers the 18mR/hr setpoint (or if the candidate confuses U1 and U2 setpoints) and then only 2 SBGT systems are expected to start since 2 monitors are above 18 mR/hr and 2 monitors are below 18 mR/hr.

B. Incorrect, see description above.
Plausible if the candidate assumes that Refuel Floor Ventilation radiation only aligns SBGT to the Refuel Floor.

C. Incorrect, see description above.
Plausible since the candidate may confuse the rad monitors and their setpoints. If the candidate remembers the 18mR/hr setpoint (or if the candidate confuses U1 and U2 setpoints) and then only 2 SBGT systems are expected to start since 2 monitors are above 18 mR/hr and 2 monitors are below 18 mR/hr.

D. Correct, see description above.

SYSTEM: 261000 Standby Gas Treatment System

A2. Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
(CFR: 41.5 / 45.6)

A2.12 High fuel pool ventilation radiation: Plant-Specific. . . . . RO 3.2 / SRO 3.4

Reference(s) used to develop this question:

34AB-T22-003-2, "Secondary Containment Control"
T46-SBGT-LP-03001, "Standby Gas Treatment System" 030.006.a.01

Reference(s) provided to the student:
None

29. 262001K4.02 001/2/1/R22 BREAKER TRIP/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is at 50% power.

o 4160 VAC buses "2A" and "2B" are all being powered from their alternate source due to maintenance activities.

- The "2G" 4160VAC normal supply breaker trips due to an internal fault.

Which ONE of the following predicts the status of the plant TWO MINUTES after the supply breaker trips?

The Reactor will have a _____.

A. full scram due to loss of Main Condenser Vacuum

B. full scram due to loss of Condensate and Condensate Booster Pumps

C. half scram (only); however, the plant will continue to operate at 50% power with no significant threats to plant operation

D. half scram (only); however, manual action will be required to prevent Turbine Building loads, cooled by PSW, from over-heating

A. **Correct**
When the alternate breaker closes in for the Emergency Bus, 4160 VAC bus 2A and 2B de-energize. This results in a loss of Main Circ Water pumps, which results in very rapid loss of main condenser vacuum. The loss of vacuum results in a main turbine trip which causes a full reactor scram since power is > 27.6%.

B. **Incorrect**
4160 C&D will remain energized following the turbine trip since they are already on alternate. **Plausible** since the student must know the correct power supplies to the Condensate and Condensate Booster Pumps in order to eliminate this distractor. The candidate(s) will choose this distractor if they think 4160VAC "A" and "B" supply power to these pumps,

C. **Incorrect**
A full reactor scram will occur due to loss of condenser vacuum. **Plausible**; if the candidate does not know the interlock which de-energizes 4160VAC bus A&B if an emergency bus is on alternate, this will look like the correct answer.

D. **Incorrect**
A full reactor scram will occur due to loss of condenser vacuum. **Plausible**; if the candidate does not know the interlock which de-energizes 4160VAC bus A&B if an emergency is on alternate, this will look like a possible correct answer. The PSW loads overheating is a distraction implying the Turbine Building isolation valves close due to a temporary loss of power to the 4160 VAC bus (if the bus swaps to alternate, the valves do not actually go closed. If the bus de-energizes and remains de-energized, the valves actually remain open since they are MOVs)

SYSTEM: 262001 A.C. Electrical Distribution

K4. Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K4.02 Circuit breaker automatic trips . . . . . RO 2.9 / SRO 3.3

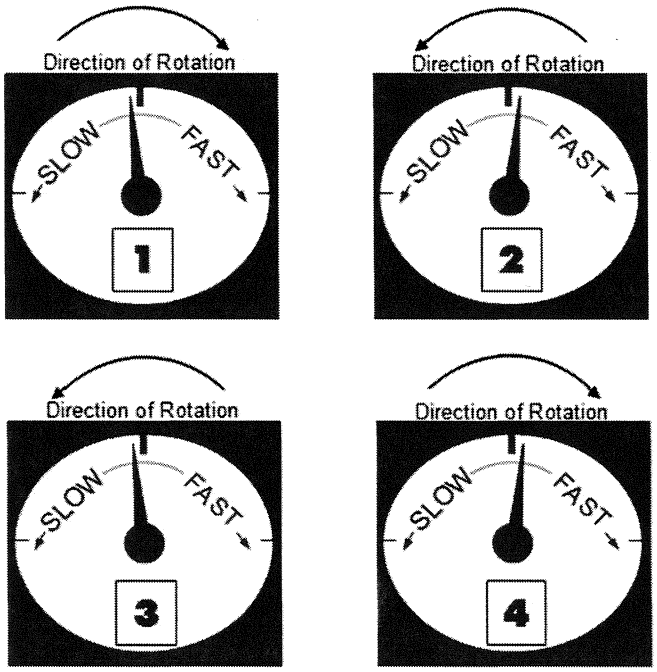
Reference(s) used to develop this question:

34AB-R22-002-2, "Loss of 4160V Bus Interlocks"
R22-4160 VAC-LP-02702, "4160 VAC" EO 027.009.a.03
34SO-R22-001-2, "4160 VAC System"

Reference(s) provided to the student:
None

30. 262001K4.05 001/2/1/EDG , SYNCHROSCOPE/BANK MOD/FUND/HT2009-301/RO/ELJ/CME/

An operator is preparing to close the "1A" Emergency Diesel Generator (EDG) output breaker during the performance of 34SV-R43-001-1, "Diesel Generator 1A Monthly Test".



IAW 34SV-R43-001-1, which synchroscope (1, 2, 3 or 4) indicates that the required conditions are met to manually close the breaker based on both of the following?

- o Direction of rotation
- o Pointer placement

A. 1

B. 2

C. 3

D. 4

A. **Correct**
The procedure requirements are: rotation in the clockwise (fast) direction and the synchroscope indicates 2 minutes to 12.

B. **Incorrect**
Direction of rotation is not correct. The pointer is on the wrong side of the 12:00 position. **Plausible** because these are the correct conditions for closing the normal or alternate supply breaker when the EDG is supplying power to the emergency bus.

C. **Incorrect** because the direction of rotation is incorrect. Direction of rotation is **plausible** because it is the required direction for closing the normal or alternate supply breaker when the EDG is supplying power to the emergency bus. The pointer placement is in the correct position.

D. **Incorrect** because the pointer is in the wrong position (past the 12:00 position). Direction of rotation is correct. Pointer indication is **plausible** because it is the correct position for closing the normal or alternate supply breaker when the EDG is supplying power to the emergency bus.

SYSTEM: 262001 A.C. Electrical Distribution

K4. Knowledge of A.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K4.05 Paralleling of A.C. sources (synchroscope) . . . . . RO 3.4 / SRO 3.6

Reference(s) used to develop this question:

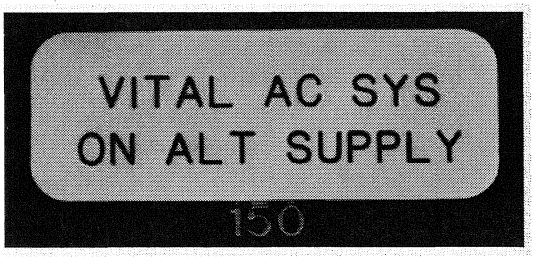
34SV-R43-001-1, Diesel Generator 1A Monthly Test
R43-EDG-LP-02801, "Emergency Diesel Generators" 028.023.a.11

Reference(s) provided to the student:
None

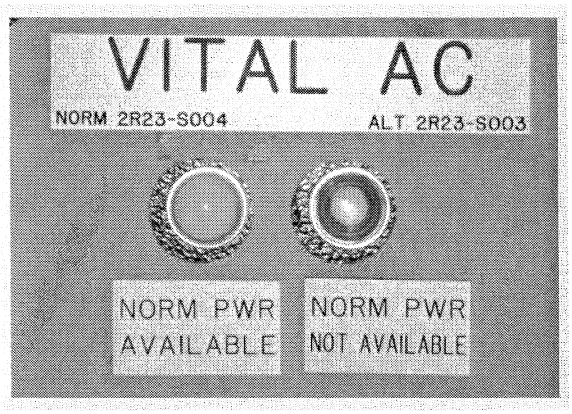
31. 262002A3.01 001/2/1/VITAL AC/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 86% of rated power.

An alarm is received in the main Control Room. As you scan the control panels you notice the following alarm and indications:



(Alarming)



(Amber Light Illuminated)

Using these indications, determine the power source which is supplying the Vital AC Bus:

A. 600 VAC Bus 2C through the Vital AC Static Inverter.

B. 600 VAC Bus 2D through the Vital AC Static Inverter..

C. 600 VAC Bus 2C, through the Vital AC Essential Transformer.

D. 600 VAC Bus 2D, through the Vital AC Essential Transformer

A. **Incorrect**
600VAC bus does not supply power to the Vital AC Battery Charger that provides power to the inverter.
Plausible; candidate must know which bus supplies the battery charger to eliminate this distracter. 600VAC bus 2C does supply power to Vital AC in this instance, but not through a battery charger. It supplies power via an essential transformer and static transfer switch.

B. **Incorrect**
Normal supply is 600 VAC "2D", which is not available, per the amber and white light status.
Plausible; the candidate must know which bus is the normal supply and which is alternate. The fact that 600VAC bus "2D" is actually the normal supply through the battery chargers, static inverter, etc.

C. **Correct**

D. **Incorrect**
The 600VAC bus "2D" does not supply the Essential transformer.
Plausible the candidate must know which bus is the normal supply and which is alternate. 600VAC bus "2D" is actually the normal supply which does not go through the ess. xfrmr.

SYSTEM: 262002 Uninterruptable Power Supply (A.C./D.C.)

A3. Ability to monitor automatic operations of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) including: (CFR: 41.7 / 45.7)

A3.01 Transfer from preferred to alternate source . . . . . RO 2.8 / SRO 3.1

Reference(s) used to develop this question:

34AR-651-150-2, Vital AC Sys on Alt Supply ARP
R25-ELECT-LP-02705 Vital AC lesson plan EO 200.020.A.05 Given a set of conditions, select the correct power supply (if any) to the Vital AC bus.

Reference(s) provided to the student:
None

32. 263000A2.02 001/2/1/DC VENTILATION/BANK/FUND/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 100% power.

- The Control Building Ventilation System normal ventilation is lost.

IAW 34AB-T41-001-2, "Loss Of ECCS, MCREC Or Area Ventilation System(s)", which ONE of the following actions is required to protect the Station Service DC Electrical Distribution System?

A. Start the emergency battery room exhaust fans to reduce the potential for buildup of combustible gases.

B. Add electrolyte to all 125/250 VDC battery cells due to the expected increase in the rate of electrolyte evaporation.

C. Rotate the battery chargers so that in-service time for each battery charger is minimized, thereby reducing battery charger heatup.

D. Reduce DC loads in the plant to minimize the current being drawn from the 125/250 VDC batteries, thereby reducing battery heatup.

A. **Correct**

B. **Incorrect**
No guidance exists in the procedure to do add electrolyte ; however, the concept that higher temperatures will result in accelerated evaporation of water in the battery is **plausible/true**, so the requirement to add electrolyte is **plausible.**

C. **Incorrect**
No guidance exists in the procedure to rotate battery chargers; however, it is **plausible** to think that electrical equipment will heat up over time and require down time to cool off. We do rotate some equipment when cooling is lost (loss of PSW procedure we rotate Cond/Cond Booster Pumps based on high temperature alarms)

D. **Incorrect**
Normally, DC is supplied by the battery chargers rather than the batteries. Nothing in conditions given suggest a change from this situation. **Plausible** if the candidate assumes the batteries, rather than the battery chargers, normally supply DC loads.

SYSTEM: 263000 D.C. Electrical Distribution

A2. Ability to (a) predict the impacts of the following on the D.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.02 Loss of ventilation during charging . . . . . RO 2.6 / SRO 2.9

Reference(s) used to develop this question:

34AB-T41-001-2, "Loss Of ECCS, MCREC Or Area Ventilation System(s)"
Z41-CBHVAC-LP-03703, "Control Building HVAC"
34AB-P41-001-2, "Loss of Plant Service Water"

Reference(s) provided to the student:
None

33. 264000K4.06 001/2/1/EDG LOAD LIMIT/BANK MOD/FUND/HT2009-301/RO/ELJ/CME/

Which ONE of the following governor control panel settings will shut off the fuel supply to a running Plant Hatch Emergency Diesel Generator?

A. Load Limit set at "0"

B. Speed Droop set at "0"

C. Load Limit set at "10"

D. Speed Droop set at "100"

A. **Correct**

B. **Incorrect,** adjusting the speed droop to 0 results in the diesel increasing the position of the fuel racks when additional load has been placed on the EDG, thereby not affecting the shutdown of the diesel.
Plausible as this is one of the actual controls on the EDG ands is adjusted during surveillances from 0 to 50 but an infrequent manipulation.

C. **Incorrect,** adjusting this knob to 10 allows the diesel to operate at 100% fuel rack position.
Plausible since it is the correct knob for shutting down the EDG, but not using terms like fully upscale. The candidate must remember that 10 equals full scale and 100% speed. Also this knob is seldom operated so familiarity with how it works requires significant knowledge of the EDG controls. Additionally, max load limit may be interpreted as the most limiting or zero fuel.

D. **Incorrect,** adjusting this knob to 100 allows the diesel to slow by 7% speed before changing fuel supply for added electrical loading.
Plausible since it is a controller on the governor and is periodically adjusted during EDG runs but not to this value. Also this knob is seldom operated so familiarity with how it works requires significant knowledge of the EDG controls.

SYSTEM: 264000 Emergency Generators (Diesel/Jet)

K4. Knowledge of EMERGENCY GENERATORS (DIESEL/JET) design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

K4.06 Governor control . . . . RO 2.6 / SRO 2.7

Reference(s) used to develop this question:

R43-EDG-LP-02801, "Emergency Diesel Generators" 028.022.a.03 , Section V, Control features and interlocks

Reference(s) provided to the student:
None

34. 264000K5.06 001/2/1/EDG LOAD SEQUENCE/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Following a LOCA/LOSP on **Unit 1**, the following sequence occurs:

- o The "1A" and "1B" EDGs start
- o The "1C" EDG fails to start

When the EDG output breakers close, pumps start in this sequence:

- o Time = 0 sec "1A" Core Spray pump starts
- o Time = 0.5 sec "1C" RHR pump starts
- o Time = 10 sec "1D" PSW pump starts
- o Time = 12 sec "1A" and "1D" RHR pumps start
- o Time = 22 sec "1A" PSW pump starts

Which ONE of the following states the status of the load sequence and operational implication for both of the statements below?

- o The "1A" EDG load sequence was ____(1)____.
- o The "1B" EDG load sequence was ____(2)____.

A. (1) correct, operation is as designed.

(2) correct, operation is as designed.

B. (1) correct, operation is as designed.

(2) NOT correct and there is a high probability of experiencing an overload condition.

C. (1) NOT correct and there is a high probability of experiencing an overload condition.

(2) correct, operation is as designed.

D. (1) NOT correct; however, there should be no adverse consequences to EDG operation.

(2) NOT correct; however, there should be no adverse consequences to EDG operation.

Actual correct loading sequence:
Time = 0 sec Both CS pumps
Time = 0.5 sec RHR pump "1C"
Time = 12 sec RHR pumps "1A", "1B" and "1D" RHR start
Time = 22 sec PSW Pumps "1A"and "1B"
Time = 32 sec PSW pump "1C" starts if "1A" did not
Time = 37 sec PSW pump "1D" starts if "1B" did not

The above sequence assumes all 3 EDGs start. In this question the "1C" EDG did not start, so the pumps on "1G" 4160VAC bus will not run ("1B" RHR, "1B" CS & "1B" PSW)

A. **Incorrect;** See correct loading sequence above.
1st part is correct
2nd part is incorrect (loading sequence is not correct)
Plausible if the candidate does not know the EDG loading sequence timers. Changing times associated with the pump starts will make this a correct answer

B. **Correct;** See correct loading sequence above.

C. **Incorrect;** See correct loading sequence above.
1st part is incorrect (loading sequence is correct)
2nd part is incorrect (loading sequence is not correct)
Plausible if the candidate does not know the EDG loading sequence timers. Changing times associated with the pump starts will make this a correct answer.

D. **Incorrect;** See correct loading sequence above.
1st part is incorrect (loading sequence is correct)
2nd part is incorrect (loading sequence is not correct)
Plausible if the candidate does not know the EDG loading sequence timers. Changing times associated with the pump start will make this a correct answer, i.e. making the times longer than designed would make the sequence still incorrect, yet allow additional time for motor starting currents to subside.

SYSTEM: 264000 Emergency Generators (Diesel/Jet)

K5. Knowledge of the operational implications of the following concepts as they apply to EMERGENCY GENERATORS (DIESEL/JET) : (CFR: 41.5 / 45.3)

K5.06 Load sequencing . . . . . RO 3.4 / SRO 3.5

Reference(s) used to develop this question:

34AB-R22-002-1, Loss of 4160V Emergency Bus, Attachment 3
R43-EDG-LP-02801, Emergency Diesel Generators lesson plan, Table 3
EO 028.025.A.02 Given a list, select the proper loading sequence for any 4160 VAC bus after the D/G starts and the following conditions exist; a. LOSP b. LOCA/LOSP

Reference(s) provided to the student:
None

35. 268000A2.01 001/2/2/DW LEAK TS/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 experiences a rupture on the "2A" Core Spray (CS) suction piping in the Torus Area of the Reactor Building.

The following annunciators are alarming (not a complete list of all alarming annunciators)

- o **TORUS S-E AREA INSTR SUMP LVL HIGH** (657-089-2)
- o **RB S-E DIAGONAL FLOOR DRN SUMP LEVEL HIGH-HIGH** (657-034-2)

The following valve alignment exists:

- o 2T45-F002 "Torus N-E and S-E Outboard Sump Isol Valve" is open
- o 2T45-F003 "Torus N-E and S-E Inboard Sump Isol Valve" is open

Which ONE of the following completes BOTH of these statements?

If NO operator action is taken, the flow of water to the Radwaste facility ___(1)___ automatically isolate as Torus area water levels continue to increase.

IAW 34AR-657-034-2, "RB S-E DIAGONAL FLOOR DRN SUMP LEVEL HIGH-HIGH", 2T45-F002, "Torus N-E and S-E Outboard Sump Isol Valve" is required to be ___(2)___.

A. (1) will

(2) open

B. (1). will

(2) closed

C. (1) will NOT

(2) open

D. (1) will NOT

(2) closed

Description: As water leaks into the Torus room area, it enters instrument sumps and then flows into the diagonal floor drain sump. There are isolation valves located between the diagonal floor drain sumps and the torus room sumps. These valves automatically close if level in the floor drain sumps reaches the High-High level setpoint, the isolation valves automatically close to prevent flooding the diagonal. When floor drain sump level decreases, the valves automatically open to process more water. If the diagonal floor drain sump pumps can keep up with the drainage coming from the Torus room area, the isolation valves between the diagonal and the torus room area remain open. If the Torus area instrument sump High level alarm is received, its ARP requires that the isolation valves be verified to be open.
In the case of this question, the isolation valves fail to auto close. The Diagonal Floor Drain sump High-High level ARP requires the valves be confirmed closed.

A. **Incorrect,** see description above
1st part is **plausible** since the candidate is required to know which sump level corresponds to the demand to close the isolation valve. If the candidate assumes the isolation occurs at high-high-high, then this will appear to be correct..
2nd part is **plausible** since the torus high level instrument sumps ARP requires the valves to be open to allow drainage of water from the torus area to the diagonal floor drain sumps.

B. Incorrect; see description above
1st part is **plausible** since the candidate is required to know which sump level corresponds to the demand to close the isolation valve. If the candidate assumes the isolation occurs at high-high-high, then this will appear to be correct.
2nd part is correct

C. Incorrect; see description above
1st part is correct
2nd part is **plausible** since the torus high level instrument sump ARP requires the valves to be open to allow drainage of water from the torus area to the diagonal floor drain sumps.

D. Correct; see description above

SYSTEM: 268000 Radwaste

A2. Ability to (a) predict the impacts of the following on the RADWASTE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6)

A2.01 System rupture . . . . . RO 2.9 / SRO 3.5

Reference(s) used to develop this question:

T22-SC-LP-01301 "Secondary Containment" Simplified diagrams (Fig 2 & 4) and Terminal Objective 201.078.b
TORUS S-E AREA INSTR SUMP LVL HIGH (657-089-2)
RB S-E DIAGONAL FLOOR DRN SUMP LEVEL HIGH-HIGH (657-034-2)

Reference(s) provided to the student:
None

36. 286000A1.01 001/2/2/FIRE PROTECTION/BANK/FUND/HT2009-301/RO/ELJ/CME/

A plant fire protection sprinkler system actuation results in a fire main header pressure of 98 psig.

Which ONE of the following predicts the fire pump response?

A. Only the electric fire pump starts.

B. Only the "A" and "B" diesel fire pumps start.

C. Only the electric fire pump and the "A" diesel fire pump starts.

D. "A" and "B" diesel fire pumps and the electric fire pump starts.

Actual Pump start pressures:

- o Electrical fire pump 1X43-C001 starts at 110 psig
- o Diesel fire pump 1X43-C002A starts at 100 psig
- o Diesel fire pump 1X43-C002B starts at 90 psig

A. **Incorrect;** see actual pump start pressures listed above.
Plausibility, changing the system pressure to >100 psig and <110 psig would make this a correct answer.

B. **Incorrect;** see actual Pump start pressures listed above
Plausibility, if the candidate assumes the diesel fire pumps start before the electric fire pump.

C. **Correct;** see Actual Pump start pressures listed above

D. **Incorrect;** see actual pump start pressures listed above.
Plausibility, changing the system pressure to less than 90 psig would make this a correct answer.

SYSTEM: 286000 Fire Protection System

Al. Ability to predict and/or monitor changes in parameters associated with operating the FIREPROTECTION SYSTEM controls including: (CFR: 41.5 / 45.5)

A1.01 System pressure . . . . . RO 2.9 / SRO 2.9

References used to develop this question:

34SO-X43-001-1, Fire Pumps Operating Procedure
X43-FPS-LP-03601, Fire Protection lesson plan
EO 036.020.B.13 Given a fire has started, describe the starting sequence for the three fire pumps and the pressure when they start

Reference(s) provided to the student:
None

37. 295001AK2.03 001/1/1/RECIRC/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 is operating at 63% power with both Reactor Feedwater pumps in service.

- o Recirculation pump speed is at 63%
- o Steam flow is 63%
- o The "1A" Condensate Booster pump (CBP) trips
- o The "1C" CBP automatically starts.
- o The "1A" Reactor Feedwater pump trips due to low suction pressure.
- o The "1B" Reactor Feedwater pump continues to operate.
- o +23 inches is the lowest Reactor Water Level during the transient

With no operator action, the final speed of the Reactor Recirculation pumps will be _____.

A. 33%

B. 53%

C. 61%

D. 63%

Description: The Reactor Recirc pumps run back to 33% if the following conditions are met, (steam flow is above 65% or RWL < 32") AND the RFPs receive a TMR trip AND at least one RFP has <20% flow. In this case, steam flow is initially below 65%; therefore the 33% runback will not occur until the RFP trips and RWL goes below 32 inches.

The Recirc Pumps have a variable runback feature, when RWL goes below 30", Recirc Pump speed is reduced by 6.7%/inch, down to a low of 33%.

A. **Correct**, see description above

B. **Incorrect,** see description above
Plausible if the candidate calculates the runback based on the variable Recirc Runback 23" RWL is a 7 inch drop from +30" therefore a runback signal of 6.7%/inch*7 inches=47% (100%-47%=53% speed.) This runback did actually occur, but is not the most limiting.

C. **Incorrect**, see description above
Plausible since the Recirc Pumps do run back to 61% following a CBP or RFP low suction pressure conditions (following a 5 or 10 sec time delay, respectively). This runback may occur, however it will not be the most limiting runback.

D. **Incorrect**, see description above
Plausible if the candidate considers that steam flow below 65% is required for this runback to occur, without taking low RWL (below 32") into account. This is a popular misconception.

APE: 295001 Partial or Complete Loss of Forced Core Flow Circulation

AK2. Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following: (CFR: 41.7 / 45.8)

AK2.03 Reactor water level..... RO 3.6 / SRO 3.7

Reference(s) used to develop this question:

B31-RRS-LP-00401, "N21-CNDFW-LP-00201" EO 004.001.a.07
34SO-B31-001-1, "Reactor Recirculation System" step 5.2.17 "Recirc Speed Limiter Summary"

Reference(s) provided to the student:
None

38. 295003G2.4.34 001/1/1/EDG/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 is experiencing a Loss of Offsite Power (LOSP).

The following conditions exist:

- o The "1A" Emergency Diesel Generator (EDG) failed to start
- o An operator is dispatched to locally start the "1A" EDG
- o All "Control Power On" lights on 1R43-P001A are illuminated

IAW with 34AB-R43-001-1, Diesel Generator Recovery, which ONE of the following describes the requirements for starting the "1A" EDG locally AND flashing the generator field?

The "1A" EDG Control Switch must be placed in the __(1)__ position prior to depressing the "START" push-button.

The Control Switch must be in the __(2)__ position to enable the Generator Field Flash circuit.

A. (1) "REMOTE"

(2) "AT ENG"

B. (1) "REMOTE"

(2) "REMOTE"

C. (1) "AT ENG"

(2) "AT ENG"

D. (1) "AT ENG"

(2) "REMOTE"

Description: The EDG start logic requires the EDG Control Switch to be placed in the "At Eng" position for the local start switch to be used to start the EDG. In order to "flash" the generator field, the switch must be placed in the "Remote" position.

Plausibility is based on the candidate recalling the design features associated with the EDG start logic.

A. **Incorrect**; see description above
Both parts are incorrect
1st part is **plausible** if the candidate assumes the correct position for this task in "Remote"
2nd part is **plausible** if the candidate assumes the correct position for this task in "At Eng"

B. **Incorrect**; see description above
1st part is **plausible** if the candidate assumes the correct position for this task in "Remote"
2nd part is correct

C. **Incorrect**; see description above
1st part is correct
2nd part is **plausible** if the candidate assumes the correct position for this task in "At Eng".
The switch is placed back in the At-Eng position after the field has been flashed

D. **Correct**; see description above

APE: 295003 Partial or Complete Loss of A.C. Power

2.4 Emergency Procedures / Plan

2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13)
IMPORTANCE RO 4.2 / SRO 4.1

Reference(s) used to develop this question:

34AB-R43-001-1
R43-EDG-LP-02801 EDG lesson plan
EO 027.051.a.01
EO 028.006.a.02, 028.024.a.01

Reference(s) provided to the student:
None

39. 295004AA1.03 001/1/1/DC ELECTRICAL/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 was operating at 100% power.

- o A loss of 2R22-S016 (125/250VDC "A") occurs.
- o Subsequently a Main Turbine trip occurs

Which ONE of the following completes both of these statements?

- The Main Generator PCBs __(1)__.

- When the PCBs open/are opened all station service 4160 VAC buses __(2)__ automatically transfer to their alternate source.

A. (1) will automatically open

(2) will

B. (1) must be manually opened IAW 34AB-R22-001-2, "Loss of DC Buses"

(2) will

C. (1) will automatically open

(2) will NOT

D. (1) must be manually opened IAW 34AB-R22-001-2, "Loss of DC Buses"

(2) will NOT

Description: If a Main Turbine trip occurs, the Main Generator output breakers will NOT auto open. 4160v buses will NOT transfer to startup supply.

Plausibility is based on whether the candidate remembers the impact of a loss of DC on both the main generator PCBs for a turbine trip and whether the station service buses will auto transfer in this condition.

A. **Incorrect**; see description above.
Plausible since, normally, this is the correct operation of these breakers. This would also be the correct answer if the S017 had been lost rather than S016

B. **Incorrect**; see description above
Plausible since the 1st part is correct for this condition and the 2nd part is the normal operation of the station service buses.

C. **Incorrect**; see description above
Plausible since the 1st part is the normal operation of these breakers, and it would be correct if S017 had been lost rather than S016. The 2nd part is correct for this condition.

D. **Correct**; see description above

APE: 295004 Partial or Complete Loss of D.C. Power

AA1. Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: (CFR: 41.7 / 45.6)

AA1.03 A.C. electrical distribution..... RO 3.4 / SRO 3.6

References used to develop this question:

34AB-R22-001-2, "Loss of DC Buses"
EO 200.018.a.01
R42-ELECT-LP-02704 DC Electrical Lesson Plan

Reference(s) provided to the student:
None

40. 295005AK3.04 001/1/1/MT PCB/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 is at 40% power.

- o The Main Turbine automatically trips due to a problem with the lubricating oil system.

When the Main Turbine trips, the Main Generator will trip __(1)__.
This trip arrangement protects the Turbine from __(2)__.

A. (1) immediately
(2) overspeeding

B. (1) immediately
(2) overheating

C. (1) seconds later
(2) overspeeding

D. (1) seconds later
(2) overheating

Description: The generator output breakers have two types of trips, sequential and simultaneous. The sequential are associated with turbine trips and occur 3.75 seconds after the turbine trip in order to use all the steam left in the turbine after the stop valves are shut to avoid turbine damage from over-speeding. The simultaneous trips are for problems associated with the generator itself, Overcurrent, etc.

A. **Incorrect** , this does NOT result in a simultaneous trip, the concern is to prevent overspeed.
Plausible because the concern is overspeed and since the candidate must remember the type trips that cause simultaneous and sequential trip (this is confusing to most candidates).

B. **Incorrect**, this does NOT result in a simultaneous trip, it is a sequential trip to ensure an overspeed condition does not occur.
Plausible since reverse power can damage the turbine through overheating; however, overspeed protection is the over-riding concern in this condition and since the candidate must remember the type trips that cause simultaneous and sequential trip (this is confusing to most candidates).

C. **Correct,** see description above

D. **Incorrect**, because the concern is to prevent overspeed
Plausible because it is a sequential trip and since motoring of the main generator will occur for a short period of time.

APE: 295005 Main Turbine Generator Trip

AK3. Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP: (CFR: 41.5 / 45.6)

AK3.04 Main generator trip..... RO 3.2 / SRO 3.2

Reference(s) used to develop this question:

N30-MTA-LP-01701
N40-MG-LP-10002, "Main Generator"
017.006.A.09

Reference(s) provided to the student:
None

41. 295006AK1.01 001/1/1/DECAY HEAT/BANK NRC 07/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 was operating at 100% for one year when a spurious scram occurred due to surveillance testing.

The following conditions exist five minutes after the scram:

- All rods fully inserted
- MSIVs open
- Auxiliary steam loads still in service

Which ONE of the following is the expected main turbine bypass valve position and the corresponding inventory makeup that is required to maintain level constant within the normal level band?

A. 1 bypass valve will be fully open.

The required makeup is within the capacity of one CRD pump.

B. 1 bypass valve will be fully open.

The required makeup exceeds the capacity of one CRD pump.

C. 1 bypass valve will be controlling, varying between 0 - 50% open.

The required makeup is within the capacity of one CRD pump.

D. 1 bypass valve will be controlling, varying between 0 - 50% open.

The required makeup exceeds the capacity of one CRD pump.

Description: Simulator results are 0 - 50% bypass valve cycling. CRD pump capacity is ~ 120 gpm whereas actual demand is ~ 0.8 Mlbm/hr.

A. **Incorrect**; see description above.
Plausible if applicant thinks that auxiliary steam loads are sufficient to keep the bypass valves closed and does not know the post-scram steaming rate.

B. **Incorrect,** see description above.
Plausible if applicant does not understand that stem states the auxiliary steam loads are still in service.

C. **Incorrect,** see description above.
Plausible if applicant does not know the post-scram steaming rate.

D. **Correct** see description above..

APE: 295006 SCRAM

AK1. Knowledge of the operational implications of the following concepts as they apply to SCRAM: (CFR: 41.8 to 41.10)

AK1.01 Decay heat generation and removal..... RO 3.7 / SRO 3.9

References used to develop this question:

Plant specific stand-alone simulator: Mode switch to shutdown scram from 100% power resulted in the following:
1.5 minutes after scram..... main turbine trips
1.75 minutes after scram..... 2% - 50% one bypass valve cycling
3 minutes after scram..... 9% - 50% one bypass valve cycling
5 minutes after scram..... 23 - 34% one bypass valve cycling

Reference(s) provided to the student:
None

42. 295010G2.2.22 001/1/2/TS/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 90% power

- o **Unit 2** Drywell (DW) pressure is 0.5 psig.
- o At 10:00 DW pressure begins going up at 0.05 psig/minute

IAW Tech Spec Limiting Condition for Operation (LCO) 3.6.1.4, Drywell Pressure, the EARLIEST time listed that a entry into a Required Action Statement (RAS) based on DW pressure is __(1)__ and DW pressure is required to be restored to within limit no later than __(2)__ from entering the RAS.

A. (1) 10:03
(2) 15 minutes

B. (1) 10:03
(2) 1 hour

C. (1) 10:26
(2) 15 minutes

D. (1) 10:26
(2) 1 hour

Description: The TS LCO is DW pressure shall be <= 1.75 psig, and if it is not, restore it to below 1.75 psig within 1 hour. (TS 3.6.1.4)
15 minutes is the RAS for Steam Dome Pressure (TS 3.4.10)

10:03 corresponds to 0.65 psig,
10:26 corresponds to 1.8 psig which exceeds the LCO of <=1.75 psig

A. **Incorrect**; see description above.
Plausible since pressure at this time corresponds to the alarm setpoint for the Primary Containment Pressure High annunciator (34AR-603-115-2). 15 minutes is the TS RAS for steam dome pressure.

B. **Incorrect**; see description above. Second part is correct.
Plausible since pressure at this time corresponds to the alarm setpoint for the Primary Containment Pressure High annunciator (34AR-603-115-2).

C. **Incorrect**; see description above. (First part is correct).
Plausible since 15 minutes is the TS RAS for steam dome pressure.

D. Correct, see description above.

APE: 295010 High Drywell Pressure

2.2 Equipment Control

2.2.22 Knowledge of limiting conditions for operations and safety limits.
(CFR: 41.5 / 43.2 / 45.2) IMPORTANCE RO 4.0 / SRO 4.7

Reference(s) used to develop this question:

Unit 2 Tech Specs (TS 3.6.1.4 and TS 3.4.10)
T23-PC-LP-01301, Primary Containment lesson plan
300.010.a.12

Reference(s) provided to the student:
None

43. 295013AK2.01 001/1/2/RHR/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 Bulk Torus water temperature is 103°F.

IAW the 34AB-T23-003-2, "Torus Temperature Above 95°F" and 34SO-E11-010-2, "Residual Heat Removal (RHR) System", if ALL REQUIRED RHR systems are placed in Torus Cooling Mode, the TOTAL RHR cooling flow to the Torus is _____ .

A. 7,700 gpm.

B. 11,500 gpm.

C. 17, 000 gpm.

D. 23,000 gpm.

Description: IAW with 34AB-T23-001-2 requires All available torus cooling be started. The system operating procedure 34SO-E11-010-2 states that with 2 pumps operating in a loop flow should be 11, 500 gpm. With 2 loops of RHR in this mode, the max total flow is 23,000 gpm.

A. **Incorrect**, see description above.
Plausible since 7700gpm is the max torus cooling flow for one RHR pump. This would be the correct answer if torus temperature were between 95F and 100F.

B. **Incorrect**, see description above.
Plausible since 11,500 gpm is the flow for torus cooling with only one loop of RHR. The candidates will pick this answer if they only take into consideration only one loop of RHR.

C. **Incorrect**, see description above.
Plausible since 17,000 is the max RHR flow for containment spray mode per loop.

D. **Correct**, see description above.

APE: 295013 High Suppression Pool Temperature

AK2. Knowledge of the interrelations between HIGH SUPPRESSION POOL TEMPERATURE and the following: (CFR: 41.7 / 45.8)

AK2.01 Suppression pool cooling..... RO 3.6 / SRO 3.7

Reference(s) used to develop this question:

34SO-E11-010-2, "RHR System"
34AB-T23-003-2, "Torus Temperature Above 95F"
T23-PC-LP-01301, "Primary Containment" lesson plan
EO 200.040.A.01

Reference(s) provided to the student:
None

44. 295015G2.2.42 001/1/2/SBLC SLC/BANK MOD/FUND/HT2009-301/RO/CME/ELJ/

Unit 2 is in an Anticipated Transient Without a Scram (ATWS) that required a reactor water level (RWL) band of -155 inches to -185 inches.

While RWL is in this band, a Tech Spec Safety Limit Violation ___(1)___ occurred and adequate core cooling ___(2)___ exist.

A. (1) has

(2) does

B. (1) has

(2) does NOT

C. (1) has NOT

(2) does

D. (1) has NOT

(2) does NOT

Description: IAW TS, a safety limit is violated when RPV level goes below -155 inches regardless of plant conditions (ATWS/Non-ATWS). Adequate Core Cooling exists for both until RPV level goes below -185 inches (Minimum Steam Cooling RWL, i.e. steam cooling with injection) or -200 inches if there is NO injection (Minimum Zero-Injection RWL, steam cooling without injection).

A. **Correct**, see description above.

B. **Incorrect**, see description above.
1st part is correct.
2nd part incorrect; however, it is **plausible** since RWL is below TAF it may be assumed that uncovered fuel is not adequately cooled. Adequate core cooling is assured when level is above TAF, by definition.

C. **Incorrect**, see description above.
1st part is not correct; however it is **plausible** since this condition is directed by the EOPs it may be assumed the EOPs will not direct you to violate a safety limit.
2nd part is correct

D. **Incorrect**, see description above.
1st part is not correct; however it is **plausible** since this condition is directed by the EOPs it may be assumed the EOPs will not direct you to violate a safety limit.
2nd part incorrect; however it is **plausible** since RWL is below TAF it may be assumed that uncovered fuel is not adequately cooled.

APE: 295015 Incomplete SCRAM

2.2 Equipment Control

2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)
IMPORTANCE RO 3.9 SRO 4.6

Reference(s) used to develop this question:

Unit 2 Tech Specs and Bases
LP-30005, Tech Specs
EO 300.003.a.01
EOP-TERMS-LP-20304, EOP Terminology and Definitions
EO 201.093.A.08

Reference(s) provided to the student:
None

45. 295016AA2.01 001/1/1/31RS EVACUATION/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

An evacuation of the Main Control room is in progress.

- o Prior to leaving the Main Control room, RC-1 actions were performed on both units.
- o All rods did NOT fully insert on Unit 1.

All necessary actions of 31RS-OPS-001-1, "Shutdown From Outside Control Room" have been taken to establish control of **Unit 1** at the Remote Shutdown Panel.

At this time, reactor power _____.

A. can be determined by counting how many scram inlet and outlet valves are open

B. can be determined by using SPDS, in the TSC, but ONLY if power level is in the Power Range (APRMs)

C. can be determined by using SPDS, in the TSC, if power level is in EITHER the Source Range (SRMs) or the Power Range (APRMs)

D. can NOT be determined due to lack of neutron monitoring indications when operating outside in the Main Control Room

Description: SPDS can be accessed in the TSC. Immediate action for 31RS-OPS-001-1 is for the SS to send someone, preferably the STA, to the TSC to operate SPDS. RC-1 actions have been performed, SRM detectors and IRM detectors have been inserted, so power level will be indicated on the SRM's and/or APRMs.

A. **Incorrect;** Checking how many scram valves that are open is, at best, a gross estimate of control rod status. This method of checking rod position will not be accurate if there is a hydraulic lock on the SDV. The stem states that all rods did not insert, and that all the blue lights are illuminated (indicates a probable hydraulic lock of the SDV). An accurate power calculation is not possible using this method.
Plausible since 31RS-OPS-001-1 does direct operators to confirm the reactor is shutdown by visually confirming that each scram inlet and outlet valve is open, but only if SPDS is NOT available.

C. **Incorrect; 1st** part is correct (power level can still be determined using SPDS in the TSC). The second part is not correct since SRM indication is available on SPDS.
Plausible if the candidate does not consider that RC-1 being completed means that SRM detectors have been fully inserted or does not recall that SRM indication is available on SPDS.

C. **Correct**, see description above

D. **Incorrect;** SPDS can still be used to determine reactor power level (SPDS can be accessed in the TSC).
Plausible since, if SPDS were not available, this would be a correct answer (there are no SRM, IRM or APRM indications at the RSDP).

APE: 295016 Control Room Abandonment

AA2. Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT : (CFR: 41.10 / 43.5 / 45.13)

AA2.01 Reactor power..... RO 4.1* / SRO 4.1*

<u>References used to develop this question:</u>

31RS-OPS-001-1, Shutdown from outside control room
34AB-C71-001-1, Scram Procedure
X75-SPDS-LP-05601, SPDS System
056.001.A.01 Given a list of plant locations, select the locations where SPDS consoles are found

<u>Reference(s) provided to the student:</u>
None

46. 295018AK3.03 001/1/1/RBCCW/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is at 0% power after being manually scrammed from 100% power.

- o The "2A" Reactor Building Closed Cooling Water (RBCCW) pump is running
- o "2B" and "2C" RBCCW pumps can NOT be started

All actions associated with 34AB-P42-001-2, "Loss of Reactor Building Closed Cooling Water" for plant conditions in which only ONE RBCCW pump running have been performed.

The actions will ensure which ONE of the following?

A. A Control Rod Drive pump is running and is adequately cooled.

B. The Reactor Recirculation pumps are running and are adequately cooled.

C. A Fuel Pool Cooling pump is running and is protected from cavitation damage.

D. The "2A" RBCCW pump is running and is protected from pump run-out conditions.

Description: IAW 34AB-P42-001-2, Loss of Reactor Building Closed Cooling Water,
NOTE: IAW the FSAR, the following step is performed to ensure adequate RBCCW flow inside containment with a single operating RBCCW pump <u>AND</u> will prevent possible <u>pump runout</u> by isolating RBCCW to Reactor Building equipment on elevations 158 FT <u>AND</u> above.

This is done by closing isolation valves 2P42-F033 and 2P42-F034. This action isolates RBCCW flow to the Recirc MG sets and diverts any cooling to the Reactor Recirc components in the DW through an orifice (true for the CRD pumps too). This section of procedure requires the following to be tripped; RWCU (tripped and isolated), Reactor Recirc Pumps, CRD Pumps (UNLESS they are needed for Control Rod Insertion <u>AND</u> their pumping medium is less than 250°F)

The FPC pump will remain running. The procedure does not require securing the FPC pump, it does require the fuel pool temps be monitored. As Fuel Pool water temp goes up, the FPC system will get closer and closer to cavitation conditions. Fuel Pool saturation conditions can be reached in about 5 hours.

A. **Incorrect**, see description above
   Plausible since some RBCCW flow still goes to the CRD pumps through an orifice

B. **Incorrect;** see description above
   Plausible since some RBCCW flow still goes to the Reactor Recirc pumps through an orifice

C. **Incorrect;** see description above
   Plausible if the candidate thinks that RBCCW flow to the FPC heat exchanger is aligned. Also plausible because the pump will be running (procedure does not require securing this pump, it does require the fuel pool temps be monitored)

D. **Correct;** see description above

APE: 295018 Partial or Complete Loss of Component Cooling Water

AK3. Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.5 / 45.6)

AK3.03 Securing individual components (prevent equipment damage)..... RO 3.1 / SRO 3.3

Reference(s) used to develop this question:

34AB-P42-001-2, Loss of Reactor Building Closed Cooling Water
LT-LP-20201, "Introduction to Abnormal Procedures Lesson Plan
LT-LP-20201.019
P42-RBCCW-LP-00901, "RBCCW" Lesson Plan
FSAR

Reference(s) provided to the student:
None

47. 295019AA2.02 001/1/1/AIR MSIV/BANK/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 100% power.

- o 2P52-F565, "Rx Bldg Inst N2 To Non-Int Air El 185 Isol Vlv", has been tagged in the closed position.
- o Unit 2 experiences a loss of all Unit 2 Station Service Air Compressors.
- o The air cross-tie valve between Unit 1 and Unit 2 cannot be opened due to a bent stem.

Which one of the following predicts how the MSIVs will respond?

A. Inboard and Outboard MSIVs will remain OPEN

B. Inboard and Outboard MSIVs will eventually drift CLOSED

C. The Inboard MSIVs will remain OPEN

The Outboard MSIVs will eventually drift CLOSED

D. The Inboard MSIVs will eventually drift CLOSED

The Outboard MSIVs will remain OPEN

Description: The Inboard MSIVs receive pneumatic supply pressure from nitrogen, via the drywell pneumatic system. The instrument air system, normally, has no connection with the Drywell Pneumatics system. The outboard MSIVs normally receive pneumatic supply pressure from the Instrument Air system which is supplied from the Station Service Air Compressors. The backup nitrogen supply valve (2P52-F565) opens when Instrument Air pressure is lost. MSIVs drift closed on a loss of pneumatic supply pressure.

A. **Incorrect**
The outboard MSIVs will go closed. **Plausible** if the candidate thinks nitrogen is the normal supply for all MSIVs.

B. **Incorrect**
Only the outboard MSIVs will go closed based on the conditions given. **Plausibility**; the candidate assumes that air is normal pneumatic supply for all MSIVs and that nitrogen backup is not available any of the MSIVs since 2P52-F565 is tagged closed.

C. **Correct;** see description above.

D. **Incorrect**
The opposite situation is true, the inboard valves will remain open and the outboard valves will drift closed. **Plausibility**; a candidate can easily confuse which valves use nitrogen as the normal pneumatic supply and those which use nitrogen as a backup pneumatic supply.

APE: 295019 Partial or Complete Loss of Instrument Air

AA2. Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR :
(CFR: 41.10 / 43.5 / 45.13)

AA2.02 Status of safety-related instrument air system loads
(see AK2.1 - AK2.19)..... RO 3.6 / SRO 3.7

AK2.05 Main steam system.....

Reference(s) used to develop this question:

34AB-P51-001-2, "Loss of Instrument and Service Air System or Water Intrusion Into The Service Air System"
P51-P52-P70-Plant Air-LP-03501, "Plant Air lesson Plan" EO 200.025.a.05

Reference(s) provided to the student:
None

48. 295020AK1.05 001/1/2/DW COOLING/BANK MOD/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 100% power.

o Nitrogen (N_2) is being added to the Unit 2 Drywell (DW) from the Unit 1 N_2 Storage Tank

34SO-T48-002-2, "Containment Atmosphere Control and Dilution System" section 7.3.1, "Alternate Primary Containment Nitrogen Makeup From CAD loop A, Unit 1 or Unit 2 N_2 Storage Tank" is being used to add the N_2.

- o 2T48-F113, "Nitrogen to DW isolation valve" is OPEN
- o 2T48-F114, "Nitrogen to DW isolation valve" is OPEN
- o DW venting using Standby Gas Treatment is in progress
- o A fault in 2C71-P001, "RPS Power Dist Panel" results in a loss of the"2A" Reactor Protection System (RPS) bus
- o It will take 8 hours to restore the "2A" RPS bus

If the operator stationed at the 2H11-P657 panel does NOT take any action, which ONE of the following describes the operational implications for the DW?

A. ✓ N_2 addition to the DW will continue and a loss of DW cooling will eventually occur due to high DW pressure.

B. N_2 addition to the DW will continue and DW cooling will remain in operation indefinitely.

C. N_2 addition to the DW will automatically isolate and DW cooling will remain in operation indefinitely.

D. N_2 addition to the DW will automatically isolate and a simultaneous loss of DW cooling will occur.

Description: The procedure (34SO-T48-002-2) explains that 2T48-F113, 2T48-F114 will not close on a group 2 isolation and that an operator must be stationed to close the valves in the event that an isolation signal/condition actually occurs.

As a result of the loss of RPS, a group 2 signal will be generated; however, nitrogen will continue to be added to the DW. The group 2 signal will isolate the DW vent line up. As a result, DW pressure will increase and eventually result in a DW LOCA pressure signal (assuming no operator actions are taken, which the question stem states).

Typically nitrogen addition to the DW automatically isolates on a Group 2 isolation signal; however, when performing Alternate Nitrogen makeup (as in this question), it does not auto isolate.

A. **Correct**; See description above.

B. **Incorrect,** See description above. DW cooling will be lost eventually when a DW pressure LOCA signal occurs. 1st part is correct. 2nd part is **plausible** if the candidate assumes that DW venting remains in service.

C. **Incorrect,** See description above. Nitrogen addition will not auto isolate and DW cooling will eventually be lost when a DW pressure LOCA signal occurs. **Plausible** if the candidate assumes nitrogen addition auto isolates due to the Gr 2 signal generated by the loss of RPS.

D. **Incorrect,** See description above. 1st part is not correct, nitrogen addition does not auto isolate. 2nd part is not correct, DW cooling is not lost until the DW pressure signal is received (in this case). **Plausible** if the candidate assumes that DW cooling is lost as a result of a group 2 signal which is generated due to the RPS loss. DW cooling loss would actually occur if a Gr 2 signal due to DW pressure (1.85 psig) occurs.

APE: 295020 Inadvertent Containment Isolation

AK1. Knowledge of the operational implications of the following concepts as they apply to INADVERTENT CONTAINMENT ISOLATION : (CFR: 41.8 to 41.10)

AK1.05 Loss of drywell/containment cooling..... RO 3.3 / 3.6

Reference(s) used to develop this question:

34SO-T48-002-2, "Containment Atmospheric Control and Dilution Systems"
34AB-C71-002-2, "Loss of RPS"
T23-PC-LP-01301, "Primary Containment" EO 013.009.A.01

Reference(s) provided to the student:
None

49. 295021G2.4.35 001/1/1/RHR SDC/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 was in Mode 5 when a Loss of Shutdown Cooling (SDC) occurred.

An operator is dispatched to perform the following three steps from Attachment 4, "Shutdown Cooling Loop Preparation for Immediate Service", of 34AB-E11-001-2, "Loss of Shutdown Cooling".

3.28 At 130RJR19, CONFIRM 2P11-F020B OPEN, THEN OPEN Shutdown Cooling Condensate Flush Supply, 2E11-F084 AND 2E11-F083.

3.29 Monitor flow noise through the Shutdown Cooling Condensate Flush Supply, 2E11-F084 AND 2E11-F083 by listening to the pipe.

3.30 WHEN flow into the Shutdown Cooling suction pipe stops, CLOSE the Shutdown Cooling Condensate Flush Supply, 2E11-F084 AND 2E11-F083.

IAW 34AB-E11-001-2, which ONE of the following describes the purpose of steps "3.28 and 3.29" and the operational effects of continuing to step "3.30" prior to completing step "3.29".

Steps "3.28 and 3.29" ensure that ____(1)____.

The consequences of closing 2E11-F084 with flow noise still present is ____(2)____.

A. (1) the SDC suction line is completely filled with water

(2) air binding the RHR Pump when it is started

B. (1) the SDC suction line is completely filled with water

(2) decreasing RPV water level during SDC valve alignment

C. (1) adequate flow is available for flushing the line of low quality water

(2) exceeding Tech Spec water quality limits for conductivity

D. (1) adequate flow is available for flushing the line of low quality water

(2) exceeding Tech Spec water quality limits for chlorides

Description; The procedure directs the operator to close the valves 2E11-F084 AND 2E11-F083) when flow noise stops. Next, the RPV suction from recirc line is opened.

Plausibility; Plant Hatch had this very event in 1987. An upstream valve in the line-up to fill the RHR suction piping was closed. The operators did not hear any flow noise and believed the RHR piping was full. This was reported to the Control Room. When the SDC suction from the RPV was opened, RPV level dropped. This allowed an air bubble to migrate up, into to the RPV.

A. **Incorrect;** see description above. 1st part is correct. 2nd part incorrect (air bubble would migrate back to the RPV rather than the pump. **Plausible** if the candidate thinks the air bubble will be drawn to the pump when the pump is started, thereby, air binding the pump.

B. **Correct,** See description above.

C. **Incorrect;** Neither part is correct. **Plausible** since this water is relatively stagnant and there are Tech Spec limits for water quality. The Chemistry department does draw samples for water quality when normally placing the system in service.

D. **Incorrect; N**either part is correct. **Plausible** since this water is relatively stagnant and there are Tech Spec limits for water quality. The Chemistry department does draw samples for water quality when normally placing the system in service.

APE: 295021 Loss of Shutdown Cooling

2.4 Emergency Procedures / Plan

2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects. (CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 3.8 / SRO 4.0

Reference(s) used to develop this question:

LT-LP-10012, "ECCS Industry Events"
34AB-E11-001-2, "Loss of Shutdown Cooling"
E11-RHR-LP-00701, "RHR System" lesson plan, EO 200.049.A.01

Reference(s) provided to the student:
None

50. 295023AA1.04 001/1/1/RAD/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

An irradiated fuel bundle is on the **Unit 2** Refueling Bridge Main Grapple, which is in the "FULL-UP" position, and can NOT be lowered due to an equipment malfunction.

- o The Fuel Pool Transfer Canal seals deflate which causes Fuel Pool water level to decrease to its lowest possible level

Which ONE of the following describes the status of the fuel bundle that is on the Refueling Bridge Main Grapple and the Area Radiation Monitor (ARM) radiation level indications that will be observed in the Main Control Room?

Fuel Pool Water level will _____.

A. remain above the top of the Fuel Bundle and the highest indication expected on the Refuel Floor ARMs will be 4 mR/Hr

B. remain above the top of the Fuel Bundle and the highest indication expected on the Refuel Floor ARMs will be 50 mR/Hr

C. go below the top of the fuel bundle and the highest indication expected on the Refuel Floor ARMs will be 1,000 mR/Hr

D. go below the top of the fuel bundle and the highest indication expected on the Refuel Floor ARMs will be upscale

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Description; An exposed, irradiate fuel bundle will cause radiation levels of 3,000 R/Hr at 30 ft (per LT-LP-10015). A seal leak on the transfer canal seals which allows water to drop to the lowest level possible will completely uncover the fuel bundle (fuel bundles are periodically transferred through the canal to move bundles from one units pool to another). The stem states the Fuel Pool level drops to the lowest possible level.

Plausibility; a significant loss of Fuel Pool water due to deflated seals did occur at Hatch (1986) and Connecticut Yankee. There was no fuel bundle on the grapple at the time. A plant recently left a bundle hanging on a grapple for an extended period of time (days) during a holiday season.

A. **Incorrect;** see description above. **Plausible** if the candidate assumes the bundle remains covered and the rad levels on the Refuel Floor remain normal.

B. **Incorrect;** see description above. **Plausible** if the candidate assumes the bundle remains covered but some increase in rad levels occur due to lowered Fuel Pool water level. 50 mR/Hr is max normal rad levels on the Refuel Floor.

C. **Incorrect;** see description above. **Plausible** if the candidate assumes the bundle becomes partially uncovered with a significant increase in rad levels due to lowered Fuel Pool water level. 1000 mR/Hr is max safe rad levels on the Refuel Floor.

D. **Correct;** see description above.

APE: 295023 Refueling Accidents

AA1. Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS : (CFR: 41.7 / 45.6)

AA1.04 Radiation monitoring equipment..... RO 3.4 / SRO 3.7

Reference(s) used to develop this question:

LT-LP-10015, "Refueling Industry Events" LO LT-10015.001

Reference(s) provided to the student:
None

51. 295024EA2.06 001/1/1/TORUS HEATUP/BANK NRC 05/FUND/HT2009-301/RO/ELJ/CME/

A steam line break inside containment has occurred on **Unit 1**.

- o Drywell pressure is steady at +10.5 psig.
- o Drywell or Torus sprays have NOT yet been initiated.

Which ONE of the following best describes the effect on Torus water temperature?

A. The saturation temperature of the Torus water will be lower than at normal operating parameters due to the non-condensable gases discharged to the Torus.

B.✓ The Torus water temperature will initially heat up evenly throughout the Torus due to the design of the downcomers.

C. The Torus water temperature will heat up more quickly below the area of the leak in the drywell due to more energy being distributed to the Torus in that area.

D. The Torus water average temperature is unreliable until suppression pool cooling is established to provide even mixing of the water.

Description: The steam will enter the torus via a ring header and downcomers. The ring header helps to ensure steam distribution is approximately equal throughout the torus.

Plausibility for local area heating of the torus is SRVs leaking.

A. **Incorrect**; see description above. The saturation temperature is higher due to the higher pressure. **Plausible** if the candidate confuses fundamental principles of Psat-Tsat relationship.

B. **Correct;** see description above.

C. **Incorrect;** see description above. With a steam leak the steam is distributed evenly through the downcomers. **Plausible** if the candidate does not understand the purpose/design of the ring header/ downcomers.

D. **Incorrect** ; see description above. The temperature monitors of the torus still work and the average temp is just an average of all the monitors. **Plausible** if the candidate is not familiar with how water temperature is measured and monitored in the suppression chamber.

EPE: 295024 High Drywell Pressure

EA2. Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.10 / 43.5 / 45.13)

EA2.06 Suppression pool temperature..... RO 4.1 / SRO 4.1

Reference(s) used to develop this question:

Steam Tables
T23-PC-LP-01301, "Primary Containment" lesson plan EO 200.004.A.03

Reference(s) provided to the student:
None

52. 295025G2.1.23 001/1/1/HPCI EOP-107/BANK MOD/HIGHER/HT2009-301/RO/ELJ/CME/

31EO-EOP-107-2, "ALTERNATE RPV PRESSURE CONTROL" is in progress.

- o The HPCI system is being used to control reactor pressure.
- o The HPCI flow controller setpoint is at its normal standby value

IAW 31EO-EOP-107-2, with the HPCI Turbine controller in automatic mode, to INCREASE the reactor cooldown rate (CDR), the operator is directed to throttle ___(1)___ in the ___(2)___ direction.

A.* (1) 2E41-F008, "Test to CST VLV"

(2) close

B. (1) 2E41-F008, "Test to CST VLV"

(2) open

C. (1) 2E41-F011, "Test to CST VLV"

(2) close

D. (1) 2E41-F011, "Test to CST VLV"

(2) open

Description; While HPCI is in pressure control mode with the controller in automatic, per procedure the cooldown rate (CDR) is controlled by throttling 2E41-F008, "Test to CST VLV". 31EO-EOP-107-2 specifies that throttling F008 in the closed direction will increase the CDR if the controller is in auto. If the controller is in Manual, throttling F008 will have minimal effect on CDR. In Manual the CDR is increased by increasing the controller output and decreased by reducing the controller output.

This concept has been difficult for some students to master (which direction to throttle the valve to increase CDR).

A. **Correct;** see description above.

B. **Incorrect,** 1st part is correct, 2nd is not correct, opening the valve will reduce the CDR. **Plausible** if the candidate assumes that opening the valve results in more water flow, which would require more steam flow.

C. **Incorrect,** 1st part is not correct (wrong valve). 2nd part is correct. **Plausible** if the student does not remember which valve is throttled to control CDR. The valves (F008 & F011) are in series and have the same name.

D. **Incorrect,** 1st part is not correct (wrong valve). 2nd part is not correct. **Plausible** if the candidate assumes that opening the valve results in more water flow, which would require more steam flow.

EPE: 295025 High Reactor Pressure

2.1 Conduct of Operations

2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation. (CFR: 41.10 / 43.5 / 45.2 / 45.6) IMPORTANCE RO 4.3 / SRO 4.4

Reference(s) used to develop this question:

31EO-EOP-107-2, "Alternate RPV Pressure Control"
EOP-107-LP-20318, "Alter Press Control" EO 005.015.A.04

Reference(s) provided to the student:
None

53. 295026EK2.01 001/1/1/RHR/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 is operating at 90% power.

- o A Safety Relief Valve (SRV) inadvertently opened causing Suppression Pool water temperature to increase
- o Suppression Pool water temperature reaches 97°F before operators are able to close the SRV

Which ONE of the following describes the required Residual Hear Removal (RHR) Suppression Pool Cooling alignment?

A.✓ Place only one loop of RHR in Suppression Pool cooling, and the RHR heat exchanger is required to be isolated prior to starting the RHR pump.

B. Place only one loop of RHR in Suppression Pool cooling, and the RHR heat exchanger is NOT required to be isolated prior to starting the RHR pump.

C. Place all available RHR loops in Suppression Pool cooling, and the RHR heat exchanger is required to be isolated prior to starting the RHR pumps.

D. Place all available RHR loops in Suppression Pool cooling, and the RHR heat exchanger is NOT required to be isolated prior to starting the RHR pump.

Description; Between 95F and 100F, 34AB-T23-003-1 requires one loop of RHR to be placed in service. When below 100F the RHR HX is required to be isolated. Above 100F, all available RHR pumps except for pumps required for adequate core cooling. Above 100F, the RHR heat exchanger is not required to be isolated prior to starting RHR pumps since it is being placed in service to support the EOPs.

A. **Correct**; see description above

B. **Incorrect**; see description above. 1st part is correct. 2nd part is not correct since conditions to enter the EOPs have not yet been reached. **Plausible** of the candidate does not recall the difference between RHR alignment requirements above and below 100F.

C. **Incorrect**; see description above. 1st part is not correct. This is only required if operation is being directed by the EOPs. **Plausible** of the candidate does not recall the difference between RHR alignment requirements above and below 100F. 2nd part is correct.

D. **Incorrect;** 1st part is not correct. This is only required if operation is being directed by the EOPs. **Plausible** of the candidate does not recall the difference between RHR alignment requirements above and below 100F.

EPE: 295026 Suppression Pool High Water Temperature

EK2. Knowledge of the interrelations between SUPPRESSION POOL HIGH WATER TEMPERATURE and the following: (CFR: 41.7 / 45.8)

EK2.01 Suppression pool cooling..... RO 3.9 / SRO 4.0

Reference(s) used to develop this question:

31EO-EOP-012-1, Primary Containment Control EOP flowchart
34AB-T23-003-1, Torus Temperature Above 95°F
34SO-E11-010-1, RHR System

Reference(s) provided to the student:
None

54. 295028EA1.03 001/1/1/DW COOLING/BANK MOD/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 was at 100% power when a pipe break occurred inside the Drywell (DW).

The following conditions now exist:

- o Drywell Pressure: ..... 8 psig
- o Bulk Average Drywell Temperature: .... 245°F

IAW 31EO-EOP-100-2, "Miscellaneous Emergency Overrides", the "2A" DW Chiller _____.

A. is NOT allowed to be restarted due to high DW pressure

B. is NOT allowed to be restarted due to high DW temperature

C. is allowed to be restarted. The operator must first place the LOCA override switch to "BYPASS" and then reset the 86 Lockout relay at the DW Chiller breaker.

D. is allowed to be restarted. The operator must first reset the 86 Lockout relay at the DW Chiller breaker and then place the LOCA override switch to "BYPASS".

Description: 31EO-EOP-100 prohibits performing action to restart DW chillers and coolers when DW temp >250F if the LOCA signal is due to a LOCA (i.e. can only do it if the loss of DW cooling is due to a loss of DW cooling rather than a leak if temp is above 250F). In this question temperature is below 250F, so the actions may be performed even though a pipe break inside containment has occurred.

The sequence of component manipulation is critical to restart the chiller. The LOCA override/bypass must be performed first, or the lock out relay will not reset.

A. **Incorrect;** see description above (LOCA override is allowed). **Plausible** since DW pressure is above 1.85 psig due to an actual pipe break inside the DW.

B. **Incorrect;** see description above (temp is below 250F). **Plausible** since the candidate must recall from memory the temperature limits for restarting the DW chillers during a LOCA.

C. Correct; see description above. Temperature is below 250F, the LOCA is bypassed, the lock out is reset.

D. **Incorrect;** see description above (sequence is incorrect). **Plausible** because the actions taken are the correct actions; however, the sequence is reversed. Temperature conditions allow the restart.

EPE: 295028 High Drywell Temperature

EA1. Ability to operate and/or monitor the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.7 / 45.6)

EA1.03 Drywell cooling system..... RO 3.9 / SRO 3.9

Reference(s) used to develop this question:

31EO-EOP-100-2, "Miscellaneous Emergency Overrides"
P64-PCCCW-LP-01304, "Primary Containment Cooling and Chilled Water System"
EO 013.059.A.06

Reference(s) provided to the student:
None

55. 295030EK1.02 001/1/1/NPSH/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 100% power when a Loss of Coolant Accident occurs.

- o The High Pressure Coolant Injection system is being used to control RPV water level.
- o "2A" loop of Residual Heat Removal(RHR) is placed in Suppression Pool Cooling
- o Torus level ..... 135 inches
- o "2A" RHR flow ..... 7,700 gpm
- o Torus temperature ..... 225°F
- o Suppression Chamber Pressure ..... 8 psig

Which ONE of the following completes both of these statements?

- "2A" RHR pump operation is in the ___(1)___ region of the RHR NPSH Limit graph.
- "2A" RHR flow _____(2)_____.

Reference provided

A. (1) safe

(2) must be maintained at or below its current flow rate

B. (1) safe

(2) should be increased to maximize suppression pool cooling

C. (1) unsafe

(2) changes cannot restore the pump to the safe area of the graph

D. (1) unsafe

(2) reduction to 5,000 gpm will restore operation to the safe area of the graph

Description: The correct answer to this question is dependent on analyzing the correct graph. Graph selection is determined by whether suppression pool water level is At or Above 146 inches, or Below 146 inches. Common misconception among candidates on how to use the graphs, hard copy, due to having the safe region changing as torus pressure changes.

A. **Incorrect**
1st part is incorrect (is correct on >146" graph). **Plausible** if the candidate refers to Graph 12A.
2nd part is incorrect. Actually, flow must be reduced to restore safe operation. Plausible since 7700 gpm is the procedure limit for flow on one pump. This option is dependent on the candidate referring to the wrong graph plotting operation in the safe area of the NPSH graph in the first place.

B. **Incorrect**
1st part is incorrect (is correct on >146" graph). **Plausible** if the candidate refers to Graph 12A.
2nd part is incorrect. Actually, flow must be reduced to restore safe operation. Plausible since the pump is capable of more flow and the candidate may assume this is allowed during implementation of the EOPs in which the EOPs direct all available Torus Cooling be place in service (There are conditions in the EOPs where LPCI is allowed to inject at max possible flows, without requiring flow being throttled to below 7700 GPM, e.g. adequate core cooling is not assured).

C. **Incorrect**
1st part is correct. This is only determined by analyzing EOP graph 12B
2nd part is incorrect. **Plausible** if the candidate interpolates or uses the wrong line on the graph (below 5 psig limit), this will appear to be a correct option.

D. **Correct**
1st part is correct. This is only determined by analyzing EOP graph 12B, rather than 12A.
2nd part is correct. This is only determined by analyzing EOP graph 12B, rather than 12A

Reference(s) provided to the student:
Unit 2 EOP Graph 12A, RHR Pump NPSH Limit, (Suppression Pool Water Level Below 146")
Unit 2 EOP Graph 12B, RHR Pump NPSH Limit, (Suppression Pool Water Level At or Above 146")

EPE: 295030 Low Suppression Pool Water Level

EK1. Knowledge of the operational implications of the following concepts as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.8 to 41.10)

EK1.02 Pump NPSH..... RO 3.5 / SRO 3.8

Reference(s) used to develop this question:

Unit 2 EOP Graph 12A, RHR Pump NPSH Limit, (Suppression Pool Water Level Below 146")
Unit 2 EOP Graph 12B, RHR Pump NPSH Limit, (Suppression Pool Water Level At or Above 146")
EOP-CURVES-LP-20306, "EOP Curves and Limits", EO 201.065.A.27

Reference(s) provided to the student:
Unit 2 EOP Graph 12A, RHR Pump NPSH Limit, (Suppression Pool Water Level Below 146")
Unit 2 EOP Graph 12B, RHR Pump NPSH Limit, (Suppression Pool Water Level At or Above 146")

56. 295031EA1.01 001/1/1/LP ECCS RHR CS/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 100% power.

- o A Loss of Coolant Accident occurs which results in a reactor scram.
- o Drywell pressure: 3.7 psig.
- o Reactor pressure has decreased to 285 psig.

With no additional operator action, _____ will be injecting at this time.

A. NEITHER Core Spray nor RHR

B. BOTH Core Spray and RHR

C. ONLY Core Spray

D. ONLY RHR

	Shutoff Head (Referenced on the RC & RCA EOP flowcharts)
U1 RHR	220 psig
U1 CS	280 psig
U2 RHR	220 psig
U2 CS	383 psig

Hatch U2 simulator indicates approximately 1500 gpm per CS pump at 285 psig

A. **Incorrect;** see information above. Plausible if candidate believes shutoff head for both is less than 285 psig.

B. **Incorrect;** See information above. Plausible if candidate believes shutoff head is greater than 285 psig for both

C. **Correct**; see information above.

D. **Incorrect;** See information above. Plausible if candidate believes shutoff head is greater than 285 psig for RHR but not CS.

EPE: 295031 Reactor Low Water Level

EA1. Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.7 / 45.6)

EA1.01 Low pressure coolant injection (RHR): Plant-Specific. RO 4.4* / SRO 4.4*

Reference(s) used to develop this question:

Hatch Unit 2 simulator
31EO-EOP-010-1 RC RPV Control (Non ATWS) Flowchart
31EO-EOP-010-2 RC RPV Control (Non ATWS) Flowchart
EOP-RC-LP-20308, "RPV Control (Non-ATWS)lesson plan EO 201.065.A.16

Reference(s) provided to the student:
None

57. 295032EA1.03 001/1/2/RB VENT/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 2 is operating at 100% power.

The "A" Reactor Core Isolation Cooling (RCIC) Pump Room Cooler (2T41-B004A) is operating with the control switch in the "RUN" position.

- o A Plant Service Water (PSW) system problem results in "0" PSW flow to 2T41-B004A.

When will the "2T41-B004B" RCIC Pump Room Cooler automatically start?

A. ONLY when RCIC is started.

B. ONLY due to high RCIC room temperature.

C. It requires EITHER, a high RCIC room temperature OR a RCIC start.

D. It requires BOTH, a high RCIC room temperature AND a RCIC start.

Description; The standby RCIC room cooler for RCIC will auto start when room temperature exceeds its high temperature setpoint -or- when the RCIC system is started.

A. **Incorrect**; high room temperature will also start the standby cooler. **Plausible** since this is one of the signals that causes the standby cooler to start and tests the candidates knowledge concerning these signals.

B. **Incorrect,** a RCIC start will also start the standby cooler. **Plausible** since this is one of the signals that causes the standby cooler to start and tests the candidates knowledge concerning these signals.

C. **Correct**; see description above

D. **Incorrect**, it doesn't take both signals, either one alone will start the standby cooler. **Plausible** since either of these signals will cause the standby cooler to start and tests the candidates knowledge concerning these signals.

EPE: 295032 High Secondary Containment Area Temperature

EA1. Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE : (CFR: 41.7 / 45.6)

EA1.03 Secondary containment ventilation..... RO 3.7 / SRO 3.7

Reference(s) used to develop this question:

T41-SC HVAC-LP-01303, "Secondary Containment HVAC Systems" EO 037.003.A.06
34AR-650-234-2, "SEC System Auto Initiation Signal Present"

Reference(s) provided to the student:
None

58. 295035EA2.02 001/1/2/SC DP/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 is operating at 100% power.

- o A Main Steam Line (MSL) rupture inside the Reactor Building occurs
- o The leak can NOT be isolated
- o The rupture results in a pressure spike that "blows out" the Reactor Building to Turbine Building blowout panels

As compared to the same MSL rupture in which the blowout panels remained intact, this event will result in OFFSITE release rates near the site boundary being __(1)__.

The release will be a(an) __(2)__ release.

A. (1) higher

(2) elevated

B. (1) the same

(2) elevated

C. (1) higher

(2) ground level

D. (1) the same

(2) ground level

Description; The blow out panels will allow radioactive steam to enter into the Turbine Building. The Turbine Building ventilation system discharges into the Reactor Building Ventilation system and results in a ground level type of release. A ground level release will result in a higher concentration of radiation in the vicinity of the Plant Hatch site. If the blowout panels had remained in tact, the radioactivity would have been processed via SBGT filters, with SBGT flow being sent to the Main Stack. For a release to considered as an elevated release, the release must occur via the Main Stack.

A. **Incorrect**, see description above (release is not elevated). The 1st part is correct. The 2nd part is incorrect. **Plausible** since the candidate may assume the release is elevated if it goes out the Reactor Building Ventilation Stack - or - if the candidate thinks the TB exhaust goes to the Main Stack.

B. **Incorrect,** see description above (some of the radioactive steam will not be filtered via the SBGT system filter and it is a ground level release, so rad levels near the plant site will go up). The 1st part is NOT correct. **Plausible** if the candidate assumes the release is elevated and that an elevated release will carry any radioactivity away from the plant. The 2nd part is incorrect. **Plausible** since the candidate may assume the release is elevated if it goes out the Reactor Building Ventilation Stack - or - if the candidate mistakes the TB exhaust as going to the Main Stack.

D. **Correct;** see description above.

C. **Incorrect;** see description above. 1st part is NOT correct. **Plausible** if the candidate assumes the TB ventilation flow is adequately treated, to remove radioactivity, when it passes through the RB ventilation system. 2nd part is correct.

EPE: 295035 Secondary Containment High Differential Pressure

EA2. Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE: (CFR: 41.8 to 41.10)

EA2.02 †Off-site release rate: Plant-Specific..... RO 2.8* / SRO 4.1

Reference(s) used to develop this question:

EOP-SCRR-LP-20325, "Secondary Containment Radioactive Release Control"
EO 201.082.A.01
T22-SC-LP-01302, "Secondary Containment" lesson plan
EO 200.023.B.07

Reference(s) provided to the student:
None

59. 295037EK2.12 001/1/1/RWM/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 was operating at 100% power when a reactor scram occurred.

After the scram, the following conditions exist:

- o Reactor power 6%
- o Control rod insertion is in progress IAW 31EO-EOP-103-2 "Control Rod Insertion Methods"

Without further Reactor Engineering evaluation, the operating crew can determine that the reactor will remain shutdown under ALL conditions when _____.

A. all control rods are at position "02"

B. all IRM indications are stable on Range 5

C. Standby Liquid Control (SLC) tank level is at 20%

D. two control rods at position "04" and all other control rods at position "00"

Description; The reactor is shutdown under all conditions when:

- o All control rods are position "02" -or-
- o All rods are at "00" with the exception of the highest worth control rod.-or-
- o Cold Shutdown Boron Weight has been injected into the RPV (14% tank level)

Hot Shutdown Boron Weight = 35% boron tank level.

IRMs below range 6 with no boron injection is an EOP definition for "subcritical"; however, this is not a definition for "shutdown under all conditions".

A. **Correct;** see description above.

B. **Incorrect;** see description above. (it is the definition subcritical, not shutdown under all conditions). **Plausible** if the candidate confuses the definitions for the conditions.

C. **Incorrect;** see description above, (not Cold Shutdown Boron Weight level yet). **Plausible** because the candidate must know, from memory, the value for Cold Shutdown Boron Weight to eliminate this distractor. Tank level below 14% would make this a correct answer.

D. **Incorrect**; see description above, (does not meet a definition of shutdown under all conditions). **Plausible** because this represents a much higher rod density than either all rods at position "02" or all rods at position"00" with one rod fully withdrawn.

EPE: 295037 SCRAM Condition Present and Reactor Power Above APRM Downscale or Unknown

EK2. Knowledge of the interrelations between SCRAMCONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: (CFR: 41.7 / 45.8)

EK2.12 Rod control and information system: Plant-Specific... RO 3.6 / SRO 3.8

Reference(s) used to develop this question:

C11-RWM-LP-05403
EOP-CURVES-LP-20306, "EOP Curves and Limits" EO 201.070.A.03 (CSBW)
EOP-RCA-LP-20328, "RCA Flowchart", EO 201.070.A.06 (Subcritcal)
U1 TS definitions SDM (One rod full out)

Reference(s) provided to the student:
None

60. 295038EK1.02 001/1/1/MSIV CLOSING TIME/BANK MOD/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 is operating at 18% power.

The following alarms are received soon after a cold water injection into the reactor:

- o MAIN STEAM LINE RADIATION HIGH (601-425)
- o MAIN STEAM LINE RADIATION HIGH-HIGH/INOP (603-125)
- o Main Steam Line radiation levels are 6,000 mRem/hr (increasing)
- o Main Steam Isolation Valves are OPEN
- o 1B31-F019 and 1B31-F020, "Reactor Recirculation Sample Valves" are CLOSED

IAW 34AB-B21-001-1, "Main Steam Line High Radiation or Suspected Fuel Element Failure", expected automatic actions ___(1)___ occurred.

IAW 34AB-B21-001-1, the required action(s) to protect the public from radioactive release include _____(2)_____.

A.* (1) have

(2) scramming the reactor and closing the MSIVs

B. (1) have

(2) performing a fast reactor shutdown IAW 34GO-OPS-014, "Fast Reactor Shutdown"

C. (1) have NOT

(2) scramming the reactor and closing the MSIVs

D. (1) have NOT

(2) performing a fast reactor shutdown IAW 34GO-OPS-014, "Fast Reactor Shutdown"

Description: The Recirc Sample valves auto close when MSL rad monitors exceed 2.5 x normal. The setpoint also results in MAIN STEAM LINE RADIATION HIGH-HIGH/INOP (603-125). The MSIV auto closure at this setpoint is no longer applicable at Plant Hatch. If it is suspected that the cause of this alarm is Fuel Element Failure (FEF), 34AB-B21-001-1 requires the reactor be scrammed and MSIVs closed to limit rad release to the public.

A. **Correct;** see description above.

B. **Incorrect;** see description above. 1st part is correct. 2nd part is not correct (requirement is to scram rather than perform a Fast Reactor Shutdown). **Plausible** because a Fast Reactor Shutdown is the required action if the cause of the alarms is NOT suspected to be FEF -or- only if the MSL Rad High alarm is received when FEF is suspected..

C. **Incorrect;** see description above. 1st part is not correct (Recirc Sample Valves isolated as required) **plausible** if the candidate thinks the MSIVs auto close on high MSL radiation (this used to be an auto isolation at Hatch). 2nd part is correct.

D. **Incorrect;** see description above.1st part is not correct (Recirc Sample Valves isolated as required) plausible if the candidate thinks the MSIVs auto close on high MSL radiation (this used to be an auto isolation at Hatch). 2nd part is not correct (requirement is to scram rather than perform a Fast Reactor Shutdown). **Plausible** because a Fast Reactor Shutdown is the required action if the cause of the alarms is NOT suspected to be FEF -or- only if the MSL Rad High alarm is received when FEF is suspected..

EPE: 295038 High Off-Site Release Rate

EK1. Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.8 to 41.10)

EK1.02 †Protection of the general public..... RO 4.2* / SRO 4.4*

Reference(s) used to develop this question:

B21-SLLS LP-01401, "Main Steam and Low Low Set" lesson plan EO 200.098.A.01
34AB-B21-001-1, "Main Steam Line High Radiation or Suspected Fuel Element Failure"
MAIN STEAM LINE RADIATION HIGH-HIGH/INOP (603-125)

Reference(s) provided to the student:
None

61. 300000K1.03 001/2/1/DW PNEUMATICS/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 2 is shutting down for a refueling outage.

- o Instrument Air has been aligned to the Drywell (DW) Pneumatic system
- o At 10:00 an airline break inside the DW results in an air flow rate of 50 SCFM

Air being supplied to the DW through this DW Pneumatics header ___(1)___ automatically isolate _____(2)_____.

A. (1) will

(2) immediately

B.* (1) will

(2) following a 10 minute delay

C. (1) will NOT

(2) since the Instrument Air flow is below the DW Pneumatics isolation setpoint

D. (1) will NOT

(2) since the Instrument Air to DW Pneumatics connection point is downstream of the DW Pneumatics automatic isolation valves

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Description: Air is aligned upstream of the flow sensor which causes the isolation at 30 CFM. There is a 10 minute time delay associated with the isolation.

A. **Incorrect;** see description above. 1st part is correct. 2nd part is NOT correct (there is a 10 minute time delay)

B. **Correct;** see description above.

C. **Incorrect;** see description above.1st part is not correct. 2nd part is not correct (An isolation will occur). **Plausible** since the candidate must recall the setpoint for the isolation from memory to eliminate this distractor.

D. **Incorrect;** see description above. 1st part is not correct. 2nd part is not correct. **Plausible** since the candidate must recall the piping arrangement from memory to eliminate this distractor. The candidate must also differentiate between this connection and the Emergency Nitrogen (N2 Bottles) supply for SRV operation which taps in downstream of the flow sensors which cause this isolation.

SYSTEM: 300000 Instrument Air System (IAS)

K1 Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)

K1.03 Containment air . . . . . RO 2.8 / SRO 2.9

Reference(s) used to develop this question:

P51-P52-P70-Plant Air-LP-03501, "Plant Air Systems" lesson plan EO 042.001.A.01 & EO 042.001.A.03
34AR-700-120-2 "DRWL Pneu Sys Supply Line Press Low MSIV"
34AR-700-133-2 "DRWL Pneu Sys Supply Line Press Low"

Reference(s) provided to the student:
None

62. 400000K3.01 001/2/1/PSW/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 was operating at 100% power when a scram occurred.

- o After the scram, the "1E" 4160 VAC bus de-energized and was re-energized from the 1A" Emergency Diesel Generator (EDG)
- o A total loss of Plant Service Water occurs

With NO operator action, which ONE of the following describes how the "1A" EDG will respond?

The "1A" EDG will _____.

A. continue to run until major component damage occurs

B. trip due to high jacket coolant temperature

C. trip due to high lube oil temperature

D. trip due to high crank case pressure

Description: The EDGs have several automatic trips that only function when the EDG is in Test Mode, including high jacket coolant temperature, high lube oil temperature, high crank case pressure. When the EDG is NOT in test, as is the case in this question, those automatic trips are disabled. The only trips that are in effect in this case are overspeed. low lube oil pressure, start failure and differential current.

A. **Correct;** see description above.

B. **Incorrect;** see description above. Only functions in test. Plausible since this is an actual EDG trip when the EDG is in test mode. The loss of PSW cooling will result in temperatures increasing. The candidate must recall from memory when the high temperature trip is in effect.

C. **Incorrect;** see description above. Only functions in test. Plausible since this is an actual EDG trip when the EDG is in test mode. The loss of PSW cooling will result in temperatures increasing. The candidate must recall from memory when the high temperature trip is in effect.

D. **Incorrect;** see description above. Only functions in test. Plausible since this is an actual EDG trip when the EDG is in test mode. The loss of PSW cooling will result in this pressure increasing. The candidate must recall from memory when the high pressure trip is in effect.

SYSTEM: 400000 Component Cooling Water System (CCWS)

K3. Knowledge of the effect that a loss or malfunction of the CCWS will have on the following: (CFR: 41.7 / 45.6)

K3.01 Loads cooled by CCWS . . . . . RO 2.9 / SRO 3.3

Reference(s) used to develop this question:

34AB-P41-001-1, "Loss of Plant Service Water"
34SO-R43-001-1, "Diesel Generator Standby AC System"
R43-EDG-LP-02801, "Emergency Diesel Generators" lesson plan EO 028.023.A.02,
P41-PSW-LP-03301 "PSW System" lesson plan EO 200.013.A.05
34AR-R43-104-1, "Jacket Coolant Temp High"
34AR-R43-102-1, "Lube Oil Temp High"
34AR-R43-111-1, "Crank Case Pressure"

Reference(s) provided to the student:
None

63. 500000EK3.06 001/1/2/TORUS VENT/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 has experienced a transient that results in these Primary Containment parameters:

- o Hydrogen concentration 8%
- o Oxygen concentration 7%
- o Drywell (DW) pressure .... 14 psig and slowly increasing
- o Torus level ..... 280 inches (steady)

Which ONE of the following describes the Emergency Operating Procedures (EOPs) strategy and reason(s) for the strategy for these conditions?

The EOP strategy is to ___(1)___
The reason(s) for taking this action is _____(2)_____ reduce the flammability of the Primary Containment atmosphere.

A. (1) spray the DW

(2) scrub radionuclides AND

B. (1) spray the DW

(2) ONLY to

C. (1) vent the Suppression Chamber

(2) scrub radionuclides AND

D. (1) vent the Suppression Chamber

(2) ONLY to

Description: With the concentrations of H2 and O2 given in this question, the EOPs direct the operators to:

- o Vent the Suppression Chamber if Suppression Pool water level is below 300"
- o Vent the DW if Suppression Pool water level is > 300"
- o Spray the DW and Suppression Chamber

Spraying containment will cause some of the radionuclides to be entrained in the water (scrubbing) and reduce the flammability of the Primary Containment atmosphere by adding water vapor to the atmosphere.

A. **Incorrect;** see description above. 1st part is incorrect (DW spray is not permitted when torus level is > 197.5 inches). Plausible because this is a required action for high H2 and O2 concentrations and the only reason it cant be performed is due to Suppression Chamber level. The candidate must be aware of this EOP flowchart limitation to eliminate this distractor. 2nd part is a correct statement.

B. **Incorrect;** see description above. 1st part is incorrect (DW spray is not permitted when torus level is > 197.5 inches). Plausible because this is a required action for high H2 and O2 concentrations and the only reason it cant be performed is due to Torus level. 2nd part is a partially correct statement.

C. **Correct;** see description above.

D. **Incorrect;** see description above. 1st part is correct (venting the Torus is required). 2nd part is only partially correct and states this is the ONLY reason.
Plausible because this is a required action for high H2 and O2 concentrations and since the candidate must know that there are two reasons for performing the action.

EPE: 500000 High Containment Hydrogen Concentration

EK3. Knowledge of the reasons for the following responses as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: (CFR: 41.5 / 45.6)

EK3.06 Operation of wet well vent . . . . RO 3.1 / SRO 3.7

Reference(s) used to develop this question:

EOP-PC-LP-20310, "Primary Containment Control" lesson plan EO 201.072.A.27, EO 201.072.A.25, EO 201.072.A.19
31EO-PCG-001-1, "Primary Containment Gas Control" EOP Flowchart

Reference(s) provided to the student:
None

64. 600000AK1.01 001/1/1/FIRE/NEW/FUND/HT2009-301/RO/ELJ/CME/

IAW 34AB-X43-001-2, "Fire Procedure", if a class _____ fire exists, the affected component must be de-energized.

A. A

B. B

C. C

D. D

Description: Fire classifications:

Class A - Fire that leaves an ash (wood, paper or trash)
Class B - Flammable or combustible liquids or gases)
Class C - Energized Electrical equipment fire
Class D - Combustible metals

A. **Incorrect;** see description above. It is actually an electrical fire Plausible since the candidate must recall from memory the correct fire classification

B. **Incorrect;** see description above. It is actually an electrical fire Plausible since the candidate must recall from memory the correct fire classification

C. **Correct;** see description above.

D. **Incorrect;** see description above. It is actually an electrical fire Plausible since the candidate must recall from memory the correct fire classification

APE: 600000 Plant Fire On Site

AK1 Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site:

AK1.01 Fire Classifications by type . . . . . RO 2.5 / SRO 2.8

Reference(s) used to develop this question:

Hatch GET handbook
X43-FPS-LP-03601, "Fire Protection System", lesson plan EO 200.092.A.01
34AB-X43-001-2, "Fire Procedure"

Reference(s) provided to the student:
None

65. 700000G2.4.4 001/1/1/DEGRADED GRID/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 and **Unit 2** are operating at 100% power.

Over the past hour the following has occurred:

- o Voltage in the 230 KV switchyard has decreased from 238KV to 234KV.
- o 4160 VAC bus "1E" ..... De-energized due to a ground fault
- o 4160 VAC bus "1F" and "1G" bus voltage ..... Decreased to 3840 VAC

The following **Unit 1** annunciators have been received:

- o **(4160V BUS 1E) Loss Of Offsite Power** (652-102-1)
- o **4160V BUS 1E VOLTAGE LOW** (652-122-1)
- o **4160V BUS 1F VOLTAGE LOW** (652-222-1)
- o **4160V BUS 1G VOLTAGE LOW** (652-322-1)

Select the procedure that is required to be entered and also contains actions that are required to be performed.

A. 34AB-R22-003-1, "Station Blackout" due to the Loss of Offsite Power annunciator.

B. 34AB-R22-003-1, "Station Blackout" due to the 4160 VAC bus low voltage annunciators.

C. 34AB-S11-001-0, "Operation With Degraded System Voltage" due to the 4160 VAC bus low voltage annunciators.

D. 34AB-S11-001-0, "Operation With Degraded System Voltage" due to the Loss of Offsite Power annunciator.

Description: Conditions given in the question require entry into 34AB-S11-001-0, Degraded System Voltage. 4160 VAC bus low voltage annunciator setpoint = 3867 VAC. These annunciators are entry conditions for 34AB-S11-001-0.

Entry conditions have been met for a station blackout procedure; however, since 2 Emergency Buses are energized, the procedure is immediately exited without any actions required to be performed. Requirements for entering station blackout abnormal: At least two of the three Unit 1 4160V AC Emergency busses 1E, 1F, AND 1G are de-energized, as indicated by TRIPPED breaker indications AND extinguished bus pot lights on 1H11-P652. Requirements to exit this procedure: 4.4.2 IF power is restored to more than one of the 4160V AC Emergency busses, THEN exit this procedure.

A. **Incorrect;** see description above (more than 1 Emergency Bus is energized). Plausible because the candidate must know the procedure requirements for exiting 34AB-R22-003-1 from memory in order to eliminate this distractor. Additionally the LOSP annunciator is an entry condition for 34AB-R22-003-1.

B. **Incorrect;** see description above (more than 1 Emergency Bus is energized). Plausible because the candidate must know the procedure requirements for exiting 34AB-R22-003-1 from memory in order to eliminate this distractor. Additionally the low voltage annunciators are entry conditions for 34AB-R22-003-1.

C. **Correct;** see description above.

D. **Incorrect;** see description above. 1st part is correct. 2nd part is not correct since a LOSP annunciator is not an entry condition for 34AB-S11-001-0. Plausible because a Loss of Site Power annunciator implies a degraded power condition. This alarm will be received if bus voltages continue to degrade (3,255 VAC).

APE: 700000 Generator Voltage and Electric Grid Disturbances

2.4 Emergency Procedures / Plan

2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.
(CFR: 41.10 / 43.2 / 45.6) IMPORTANCE RO 4.5 / SRO 4.7

Reference(s) used to develop this question:

34AB-S11-001-0, "Operation With Degraded System Voltage"
34AB-R22-003-1, " Station Blackout"
S11-LP-02706, "Basic Grid Operating Concepts" EO 200.116.A.01
ARP 34AR-652-122-1 "4160V Bus 1E Voltage Low"
ARP 34AR-652-102-1 "Loss of Offsite Power"

Reference(s) provided to the student:
None

66. G2.1.27 001/3/1/RWM/NEW/FUND/HT2009-301/RO/ELJ/CME/

Select the ONE choice which completes the following statement to define the primary purpose of the Rod Worth Minimizer (RWM).

- The primary purpose of the RWM is to ensure that _____.

A. fuel enthalpy does not exceed 280 cal/gm during a control rod drop accident when power is $\geq$ 29%

B. fuel enthalpy does not exceed 280 cal/gm during a control rod drop accident when reactor power is < 10%

C. the Minimum Critical Power Ratio remains greater than 1.08, while withdrawing control rods, when power is $\geq$ 29%

D. the Minimum Critical Power Ratio remains greater than 1.08, while withdrawing control rods, when reactor power is < 10%

Description: The purpose of the RWM system is to limit control rod worth such that the fuel enthalpy limit of 280 cal/gm will not be exceeded during a Control Rod Drop Accident (CRDA). TS Table 3.3.2.1-1 requires the RWM to be operable in modes 1 and 2 with thermal power <10% RTP.

This question tests the purpose of the RWM by using aspects of the RBM purpose as distracters.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control withdrawal to preclude a MCPR Safety Limit (SL) violation.
The RBM is required to be operable when power is >=29% power

On Unit 2, MCPR limit is ≥1.08 for two Recirc Pump operation.

A. **Incorrect**; see description above. 1st part is correct. 2nd part is incorrect. Plausible because >= 29% is the TS requirement for the RBM, and the candidate may confuse the requirements between the RBM and RWM.

B. **Correct**; see description above.

C. **Incorrect**; see description above. 1st part is incorrect. Plausible because the RBM does provide rod blocks to prevent MCPR from being exceeded. 2nd part is incorrect. Plausible because >= 29% is the TS requirement for the RBM, and the candidate may confuse the requirements between the RBM and RWM.

D. **Incorrect**; see description above. 1st part is incorrect. Plausible because the RBM does provide rod blocks to prevent MCPR from being exceeded due to additional rod withdrawal. 2nd part is incorrect. 2nd part is incorrect. Plausible because >= 29% is the TS requirement for the RBM, and the candidate may confuse the requirements between the RBM and RWM.

2.1 Conduct of Operations

2.1.27 Knowledge of system purpose and/or function. (CFR: 41.7)
IMPORTANCE RO 3.9 / SRO 4.0 55.41 b.6

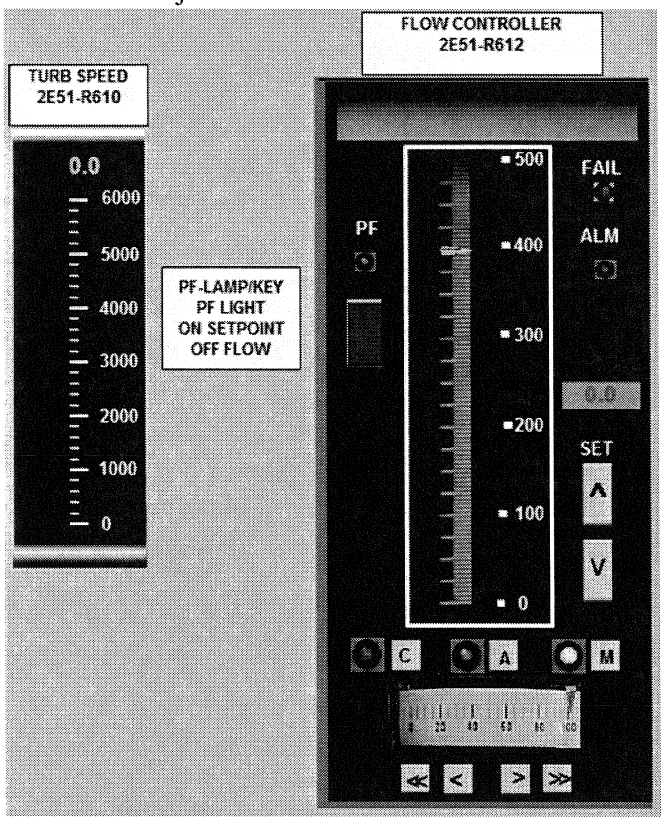
Reference(s) used to develop this question:

C11-RWM-LP-05403, Rod Worth Minimizer lesson plan. EO 001.010.D.01
C51-PRNM-LP-01203, Power Range Neutron Monitoring
TS 2.1 Safety limits
TS Table 3.3.2.1-1, Control Rod Block Instrumentation

Reference(s) provided to the student:
None

67. G2.1.31 001/3/1/RCIC RESTART/BANK MOD/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 2 RCIC has tripped with an Initiation Signal present. The trip condition has been corrected and RCIC is needed for injection.



NOTE: The light next to the "M" on the above controller is ILLUMINATED.

IAW 34SO-E51-001-2, "Reactor Core Isolation Cooling (RCIC) System", the controller __(1)__ setup for RCIC start with an initiation signal present.

This controller is located on panel __(2)__.

A. (1) IS

(2) 2H11-P601

B. (1) IS NOT

(2) 2H11-P601

C. (1) IS

(2) 2H11-P602

D. (1) IS NOT

(2) 2H11-P602

67. G2.1.31 001/3/1/RCIC RESTART/BANK MOD/HIGHER/HT2009-301/RO/ELJ/CME/

Description; Following a RCIC trip, to start RCIC for injection to the RPV, the procedure requires:

7.1.4.2 IF RCIC Initiation Signal CAN NOT BE RESET/Flow Controller Failure, THEN perform the following:

7.1.4.2.1 Transfer 2E51-R612, Turbine Controller, to MANUAL and adjust output to 50%.

7.1.4.2.2 IF necessary, RESET the Mechanical overspeed device per section 7.3.4, Mechanical Overspeed Trip Reset.

A. Incorrect, see description above. 1st part is incorrect (not at 50% output). Plausible since indicator is typically full upscale when the controller is in a standby alignment or for a restart without an initiation signal present. 2nd part is not correct (controller is on 2H11-P602). Plausible because the procedure states "Unless indicated otherwise, the following steps will be performed at panels 2H11-P601 and 2H11-P602"

B. Incorrect, see description above. 1st part is correct. 2nd part is incorrect (not at 50% output). Plausible since indicator is typically full upscale when the controller is in a standby alignment or for a restart without an initiation signal present.

C. Incorrect, see description above. 1st part is incorrect (not at 50% output). Plausible since indicator is typically full upscale when the controller is in a standby alignment or for a restart without an initiation signal present. 2nd part is correct (controller is on 2H11-P602). Plausible because the procedure states "Unless indicated otherwise, the following steps will be performed at panels 2H11-P601 and 2H11-P602"

D. Correct, see description above.

2.1 Conduct of Operations

68. G2.1.39 001/3/1/CONSERVATIVE DEC/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 is operating at 60% power.

- o The Shift Supervisor (SS) and Shift Technical Advisor (STA) are currently on tour outside the Main Control Room
- o All Oscillation Power Range Monitors (OPRM) are INOPERABLE

A Reactor Recirculation transient results in the plant operating in the Region Of Potential Instabilities.

- o The Average Power Range Monitors (APRMs) are oscillating at 6%
- o The oscillations are slowly increasing

As the Control Board Operator (CBO) in the Main Control Room, your first action IAW 34AB-C51-001-1, "Reactor Operations With Inoperable OPRM System" will be to _____.

A. scram the reactor

B. begin continuous monitoring of the APRMs

C. insert control rods after receiving STA recommendation

D. raise Reactor Recirculation flow after receiving SS permission

Description; When oscillations occur, with all OPRMs inoperable, the procedural requirement to scram IAW 34AB-C51-001-1 is:

4.1 IF at ANY time while operating in the Region of Potential Instabilities (RPI), ANY of the following conditions exist, enter 34AB-C71-001-1, Scram Procedure, AND SCRAM the Reactor:

4.1.1 Any APRM is observed to have bandwidth oscillations ≥ 5% peak to peak oscillations AND INCREASING.

Not taking action because the candidate is waiting on SS or STA permission/recommendation is an example of a decision that is not conservative (SER 23-93).

A. Correct. See description above

B. Incorrect. See description above. Plausible because the procedure provides additional guidance when a condition requiring a scram is not present; Step 4.4 IF the APRM OR LPRM flux noise increases noticeably over current OR normal values OR IF LPRM alarms are received, begin continuous monitoring of the APRMs AND LPRMs (as practical), concentrating particularly in areas where alarms are received.

C. Incorrect. See description above. Plausible because the procedure provides additional guidance when a condition requiring a scram is not present, step 4.6.2 Insert ** groups of rods to their RWM insert limit using the continuous in switch per the STA's recommendation to reduce power below the 61% load line on appropriate Attachment 1 OR 2 Power/Flow Map.

D. Incorrect. See description above. Plausible because the procedure provides additional guidance when a condition requiring a scram is not present, 4.6.1 Increase Core Flow to greater than 50% (54% if using Attachment 2) .IF possible.

2.1 Conduct of Operations

2.1.39 Knowledge of conservative decision making practices. (CFR: 41.10 / 43.5 / 45.12)
IMPORTANCE RO 3.6 / SRO 4.3

Reference(s) used to develop this question:

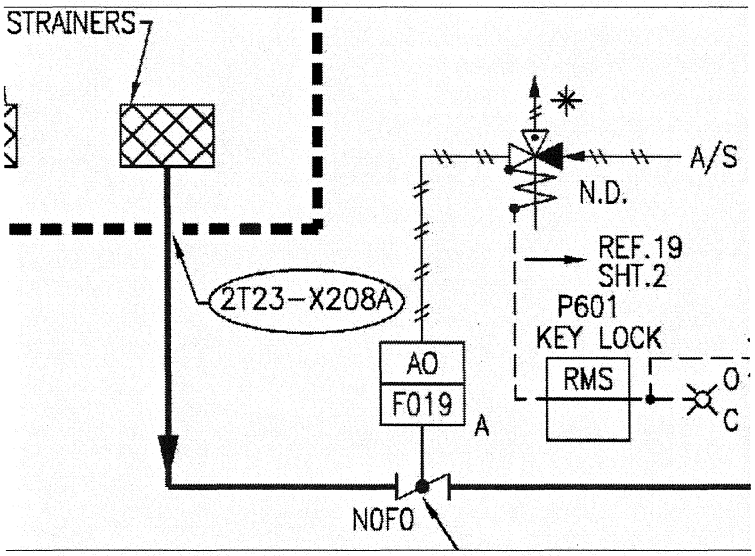
SER 23-93, "Delayed manual scram following a core flow reduction event and entry into a region of core instability"
34AB-C51-001-1, "Reactor Operations With Inoperable OPRM System"
LT-10023 IE Conservative Decision Making LO LT-10023.001

Reference(s) provided to the student:
None

69. G2.2.15 001/3/2/VENT & DRAIN/NEW/FUND/HT2009-301/RO/ELJ/CME/

The operating team is performing a tagout of the **Unit 2** Core Spray system.

- o 2E21-F019A, "Torus Suction Valve" is closed
- o 10 seconds later the 2E21-F019A's solenoid valve fuse blows (refer to drawing)



Which ONE of the following completes both of these statements?

2E21-F019A will __(1)__.

Air will be __(2)__ the 2E21-F019A actuator.

A. (1) stroke open

(2) vented off of

B. (1) stroke open

(2) supplied to

C. (1) remain closed

(2) vented off of

D. (1) remain closed

(2) supplied to

Description; 2E21-F019A fails open on a loss of air pressure. When the solenoid is de-energized, air is vented off of the actuator of F019A, which results in the valve stroking open. The candidate must use the drawing provided to determine the impact of the fuse blowing on the operation of 2E21-F019A (i.e. is the solenoid energized when the valve is open or closed, does the solenoid vent air off the actuator when the solenoid is energized or de-energized))

A. Correct; see description above.

B. Incorrect; see description above. 1st part is correct. 2nd part is not correct (air is vented off the actuator). Plausible because the candidate must use print reading skills to determine the impact and configuration of the current lineup for air and position of F019.

C. Incorrect; see description above. 1st part is NOT correct. Plausible because the candidate must use print reading skills to determine the impact and configuration of the current lineup for air and position of F019. 2nd part is correct

D. Incorrect; see description above. 1st part is NOT correct. 2nd part is NOT correct. Both are plausible because the candidate must use print reading skills to determine the impact and configuration of the current lineup for air and position of F019.

2.2 Equipment Control

2.2.15 Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc. (CFR: 41.10 / 43.3 / 45.13)
IMPORTANCE RO 3.9 / SRO 4.3

Reference(s) used to develop this question:

H26018, "Core Spray System P&ID"
H27661, "Core Spray System Elementary Diagram, sh 4 of 6"
SO-LP-00011, "Valves and Valve Operators" LO SO-00011.035

Reference(s) provided to the student:
None

70. G2.2.25 001/3/2/MCPR/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 is operating at 75% power with a Minimum Critical Power Ratio of 1.03.

If the Main Turbine were to trip AND the Main Turbine Bypass Valves fail to open, the risk of fuel damage is significantly increased due to which ONE of the following:

A. Transition Boiling

B. Stored Decay Heat

C. Zirc-Water Reaction

D. Pellet Clad Interaction

This question tests the bases for the MCPR safety limit by use of distracters composed from the bases/failure mechanism from other thermal limits. Candidates often confuse the mechanisms associated with safety/thermal limits.

A. **Correct**

B. **Incorrect;** this is the failure mechanism due to APLHGR following a LOCA. Nothing stated in the stem indicates that APLHGR is being violated. Plausible since it is one of the safety limit failure mechanisms

C. **Incorrect,** this one of the effects from elevated clad temperatures (APLHGR). Zirc Water reactor occurs when clad temperatures reach 2200°F, which is the temperature associated with APLHGR limit violation. Nothing stated in the stem indicates that APLHGR is being violated. Plausible since it is a condition associated with one of the safety limit violations.

D. **Incorrect;** this is the failure mechanism for LHGR violation and nothing stated in the stem indicates that LHGR is being violated.

2.2 Equipment Control

2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits. (CFR: 41.5 / 41.7 / 43.2) IMPORTANCE RO 3.2 / SRO 4.2

Reference(s) used to develop this question:

LT-LP-00305, "Technical Specifications" LO 300.003.A.04
TS 2.0 Safety Limits
TS Bases for safety limits B2.1.1

Reference(s) provided to the student:
None

71. G2.2.39 001/3/2/TS/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 is at 100% power, late in core life.

- o The "1B" Control Rod Drive (CRD) pump is tagged out
- o All control rods are withdrawn

The following sequence of events occur:

- o 10:00 the "1A" CRD pump trips
- o 10:05 an operator attempts to restart the "1A" CRD pump and is NOT successful
- o 10:10 the Shift Supervisor declares two CRD scram accumulators inoperable

IAW with Tech Specs 3.1.5, which ONE of the following is the EARLIEST time that the Reactor Mode Switch MUST be in the Shutdown position?

A. 10:06

B. 10:11

C. 10:31

D. 11:01

Description; When 2 or more accumulators on withdrawn control rods are inoperable with steam dome pressure greater than 900 psig and charging water header pressure cannot be restored to >940 psig within 20 minutes (TS 3.1.5) the reactor mode switch must be placed in the shutdown position immediately. The 20 minute clock for this question starts at the time when 2 accumulators are declared inoperable.

Additional plausibility and difficulty for this question comes from the fact that the candidate must answer TS RAS's from memory.

A. Incorrect; see description above. Plausible if the candidate determines that an immediate scram is required when it becomes obvious that charging water header pressure cannot be restored (cant restart the CRD pump).

B. Incorrect; see description above. Plausible because this would be a correct answer if reactor pressure were <900 psig.

C. Correct; see description above.

D. Incorrect; see description above. Plausible because the control rod is required to be declared inoperable within one hour with charging water pressure is >940 psig and the accumulators inoperable.

2.2 Equipment Control

2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems. (CFR: 41.7 / 41.10 / 43.2 / 45.13) IMPORTANCE RO 3.9 / SRO 4.5

Reference(s) used to develop this question:

Unit Two Tech Spec (TS 3.1.5)
C11-CRD-LP-00101, "CRD System" lesson plan EO 200.045.A.02

Reference(s) provided to the student:
None

72. G2.3.14 001/3/3/EOP 103/NEW/FUND/HT2009-301/RO/ELJ/CME/

Unit 1 is at 35% rated

- o The Hydrogen Injection System is placed in service in AUTOMATIC - EXTERNAL mode
- o Power is raised from 35% power to 100% power
- o At 100% power hydrogen flow rate indicates 40 SCFM

Which ONE of the following answers both of these statements?

IAW 34SO-P73-001-1, "Hydrogen and Oxygen injection and Control for HWC", hydrogen injection flow rate is __(1)__ the normal 100% power flow rate.

Radiation levels in the Condenser Bay will stabilize __(2)__ expected normal full power radiation levels.

A. (1) above
(2) at

B. (1) below
(2) at

C. (1) below
(2) below

D. (1) above
(2) above

Description;
In the Automatic - External Mode the hydrogen injection system is load following. It is normally placed in service at just above 30% Rx power. As Rx power is increased, the hydrogen flow rate is increased to the maximum amount that the controller is set for. The amount of hydrogen has varied over time from as low as 8 SCFM to as high as 45 SCFM. The plant is currently operating at 10 SCFM, but this amount is determined by the Chemistry Dept. and is not specifically listed in a procedure. There is a system shutdown on hydrogen flow of 25 SCFM at this time. There was an event at Plant Hatch: Hydrogen flow was manually lowered in Internal mode, the controller was then placed in External mode and hydrogen gas flow ramped back up to the original setpoint. This resulted in a radiation exposure to several individuals that was above the expected dose for the task. As hydrogen injection rate is increased, radiation levels increase around any area where Main Steam is piped.

A. Incorrect, radiation levels are expected to increase to above normal levels..
Plausible because flow is higher than normal, but the candidate must understand the operation of the Automatic-External mode and the effects of H2 injection on Radiation levels.

B. Incorrect, H2 flow rate is NOT below normal.
Plausible, if candidate does not understand the operations of the Automatic-External mode and the effects of H2 injection on Radiation levels or if candidate incorrectly recalls that normal flow is above 40 SCFM.

C. Incorrect, H2 flow rate is NOT below normal.
Plausible, if candidate does not understand the operations of the Automatic-External mode and the effects of H2 injection on Radiation levels or if candidate incorrectly recalls that normal flow is above 40 SCFM.

D. Correct, see above.

2.3 Radiation Control

2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)
IMPORTANCE RO 3.4 / SRO 3.8

Reference(s) used to develop this question:

34SO-P73-001-1, "Hydrogen and Oxygen injection and Control for HWC"
P73-HWCI-LP-07301, "Reactor Water Chemical Injection System", EO 026.034.A.05 and 026.034.A.04

Reference(s) provided to the student:
None

73. G2.3.7 001/3/3/RWP/NEW/FUND/HT2009-301/RO/ELJ/CME/

Which ONE of the following completes this statement?

IAW 60AC-HPX-004-0, "Radiation and Contamination Control", while signed onto the Operations GENERAL Radioactive Work Permit (RWP 09-0004), you are allowed to enter an area where _____ WITHOUT being required to sign on to a SPECIFIC RWP.

A. Very High Radiation levels exist

B. a highly contaminated system being breached

C. airborne radioactivity concentration is 0.2 DAC

D. loose surface contamination level = 300,000 dpm/100cm^2

1 _____

Description; IAW 60AC-HPX-004-0:

8.2.4.1 General RWPs are used for inspection, plant tours and general area walkdowns in **areas other than** Highly Contaminated, High Radiation, Very High Radiation or Airborne Radioactivity Areas.

Step 8.2.5 from 60AC-HPX-004, Specific RWPs are used for controlling work that is not controlled by a General RWP and occurs in a High Radiation Area, High Contamination Area or areas with changing radiological conditions. The Specific RWP will be modified on the basis of surveys. Radiological surveys **SHALL** be performed as required to support RWP authorized work.

8.2.5.3 A Specific RWP will be considered if any of the following conditions exist:

- o General area radiation levels >=100 mRem/hr
- o Airborne radioactivity concentration >30% of Derived Air Concentrations (DACs) specified in 10CFR20.
- o Loose surface contamination levels >200,000 DPM/100 cm2
- o Breach of a highly contaminated system, OR in a High or Very High Radiation Area

A. Incorrect, see description above. Plausible because the candidate must recall the requirement for General RWPs vs. Specific RWPs from memory to eliminate this distractor. Very High Rad areas require a Specific RWP.

B. Incorrect, see description above. Plausible because the candidate must recall the requirement for General RWPs vs. Specific RWPs from memory to eliminate this distractor. Areas where highly contaminated systems are being breached require a Specific RWP.

C. Correct, see description above.

D. Incorrect, see description above. Plausible because the candidate must recall the requirement for General RWPs vs. Specific RWPs from memory to eliminate this distractor. Areas with loose surface contamination level >200,000 DPM/100cm^2 require a Specific RWP.

2.3 Radiation Control

2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions. (CFR: 41.12 / 45.10)
IMPORTANCE RO 3.5 / SRO 3.6

Reference(s) used to develop this question:

60AC-HPX-004-0, "Radiation and Contamination Control"
LT-LP-30008, "Rad Control Admin Procedure" LO LT-30008.003

Reference(s) provided to the student:
None

74. G2.4.1 001/3/4/EOP ENTRY/NEW/HIGHER/HT2009-301/RO/ELJ/CME/

Unit 1 is operating at 70% power when a transient occurs.

Current plant conditions:

- o Reactor water level ..... +5 inches (lowest level reached)
- o Reactor pressure ..... 1080 psig (highest pressure reached)
- o Drywell pressure ..... 1.90 psig (highest pressure reached)

Which ONE of the following completes both of these statements?

Entry conditions have been met or exceeded ____(1)_____ Emergency Operating Procedure (EOP) flow chart(s).

IAW 34AB-C71-001-2, "Scram Procedure", performance of the RC-1, RC-2 and ___(2)___ placards are required IMMEDIATE actions.

A. (1) ONLY for the Reactor Controls (RC)

(2) TC-1

B. (1) ONLY for the Primary Containment (PC)

(2) RC-3

C. (1) for BOTH the RC and PC

(2) RC-3

D. (1) for BOTH the RC and PC

(2) TC-1

Description; EOP entry conditions are as follows:

RC EOP flowcharts are:

- o RWL < +3"
- o RPV pressure >1074 psig
- o DW pressure >1.85 psig
- o A conditions that requires reactor scram and reactor power >5%
- o A conditions that requires reactor scram and reactor power is unknown

PCC EOP flowchart:

- o Torus water level >150"
- o Torus water level <146"
- o DW pressure >1.85 psig
- o Torus water temp >100F
- o DW temp >150F
- o PC H2 concentration >1.5%

Upon entry of Scram Procedure, 34AB-C71-001, the RC-1, RC-2 and RC-3 placards are performed. TC-1 is directed to be performed as the first **subsequent** action step of the scram procedure.

A. **Incorrect,** see description above. 1st part is partially correct (entry into both RC-1 and PCC is required). 2nd part is not correct because TC-1 is not directed from an immediate action step. Plausible because the candidate must recall from memory the EOP entry condition for the EOP flowcharts in addition to recalling the Immediate Action steps from the scram procedure.

B. **Incorrect**, see description above.1st part is partially correct (entry into both RC-1 and PCC is required). Plausible because the candidate must recall from memory the EOP entry condition for the EOP flowcharts in addition to recalling the Immediate Action steps from the scram procedure. 2nd part is correct.

C. **Correct**, see description above.

D. **Incorrect**, see description above. 1st part is correct. 2nd part is not correct because TC-1 is not directed from an immediate action step. Plausible because the cantdidate must recall from memory the EOP entry condition for the EOP flowcharts in addiion to recalling the Immediate Action steps from the scram procedure.

2.4 Emergency Procedures / Plan

2.4.1 Knowledge of EOP entry conditions and immediate action steps.
(CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 4.6 / SRO 4.8

Reference(s) used to develop this question:

31EO-EOP-012-1
31EO-EOP-010-1
34AB-C71-001-1

Reference(s) provided to the student:
None

75. G2.4.9 001/3/4/LOW POWER INJECTION/NEW/HIGHER/HT2009-301/RO/CME/ELJ/

Unit 2 is shutdown with all rods fully inserted.

- o Reactor pressure ..... 35 psig
- o Main Condenser Vacuum .... "0" in. Hg. Vac
- o The "2B" loop of Residual Heat Removal (RHR) is operating in Shutdown Cooling (SDC)
- o The "2A" loop of RHR is tagged out

A Logic System Functional Test results in an inadvertent Drywell pressure Loss of Coolant Accident (LOCA) signal.

Which ONE of the following predicts the status of RHR SDC system and how reactor pressure will be controlled?

Shutdown Cooling ___(1)___.

IAW 34SO-E11-010-001-2, "RHR System" and/or 34AB-E11-001-2, "Loss of SDC", reactor pressure will be controlled by _____(2)_____.

A. (1) remains in service

(2) throttling 2E11-F048B, "RHR Heat Exchanger Bypass Valve"

B. (1) remains in service

(2) throttling 2E11-F003B, "RHR Heat Exchanger Outlet valve"

C. (1) has been lost

(2) opening the MSIVs and using the Main Turbine Bypass Valves to send steam to the Main Condenser, defeating the low vacuum trips as necessary.

D. (1) has been lost

(2) raising RWL to the Main Steam Lines (MSL) and opening the MSL drain valves, defeating the low vacuum trips as necessary.

Description; With SDC in service, when a high DW pressure signal is received SDC is lost because the RHR injection valve (2E11-F015B) will auto close; however the RHR pump remains running. Low water level (+3 inches) or high reactor pressure (138 psig) will also take RHR out of service, the difference in this case is that the F008 and F009 valves close, which causes an RHR pump trip.

With RHR in service, cooldown is controlled, per 34SO-E11-010-2, by:

7.2.3.93 WHILE maintaining RHR Loop A(B) flow rate of 6200 to 8200 GPM, establish an increase OR decrease in the cooling rate as required to obtain/maintain the desired coolant AND/OR RPV temperature(s) as specified by Shift Supervisor, by manipulating one OR a combination of the following MOVs:

7.2.3.93.1 2E11-F003A(B), Hx Outlet Vlv, (OPEN to increase cooling; CLOSE to decrease cooling).

7.2.3.93.2 2E11-F048A(B), Hx Bypass Vlv, (CLOSE to increase cooling; OPEN to decrease cooling).

7.2.3.93.3 2E11-F068A(B), Hx A(B) RHRSW Disch Vlv, (OPEN to increase cooling; CLOSE to decrease cooling).

Note: Steam pressure of 35 psig is not adequate to provide steam seals for establishing Main Condenser vacuum.

A. Incorrect, see description above. 1st part is not correct. Plausible since the RHR pump continues to run in this case. 2nd part is not correct (2E11-F015 closing will stop flow to the RPV). Plausible since this is a normal method of controlling cooldown (see above). If the candidate considers the loop to be in service, this will look to be a correct answer.

B. Incorrect, see description above. 1st part is not correct. Plausible since the RHR pump continues to run in this case. 2nd part is not correct (2E11-F015 closing will stop flow to the RPV). Plausible since this is a normal method of controlling cooldown (see above). If the candidate considers the loop to be in service, this will look to be a correct answer.

C. Incorrect, see description above. 1st part is correct. 2nd part is not correct (Bypass valves close at 7" Hg-Vac.) Plausible because using bypass to send steam to the main condenser is an alternate pressure control path. Additionally, the auto closure of the MSIVs can be overridden, per procedure, and this provides a distractor in that the candidate may confuse which low vacuum isolation is allowed to be overridden.

D. Correct, see description above. Steam seals cannot be formed on the Main Condenser so the next method in 34AB-E11-001 is to raise level and establish flow through the MSL drain valves.

2.4 Emergency Procedures / Plan

2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.
(CFR: 41.10 / 43.5 / 45.13) IMPORTANCE RO 3.8 / SRO 4.2

Reference(s) used to develop this question:

34AB-E11-001-2
34AB-E11-010-2

Reference(s) provided to the student:
None