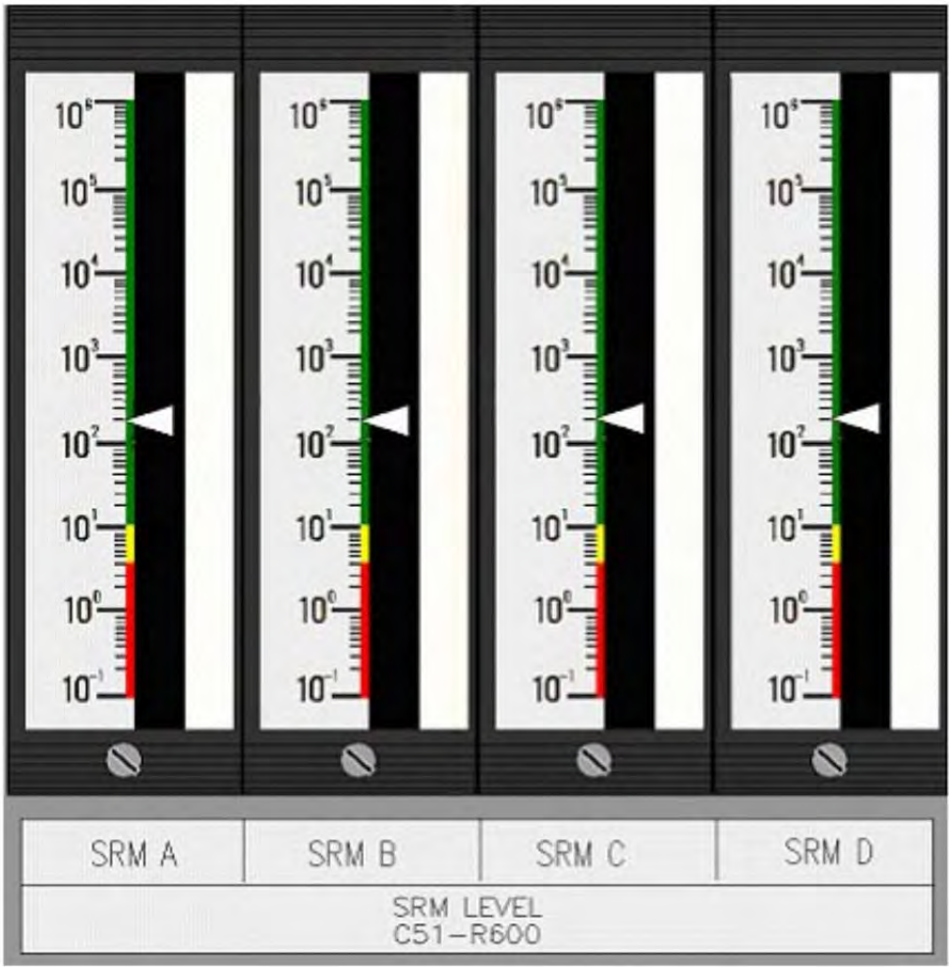
1. 201002 1

The initial SRM count rates are as observed below.



The Unit Two control room staff is ready to withdraw control rods for a reactor startup.

Which one of the following identifies when criticality is expected to be achieved IAW 0GP-02, Approach To Criticality and Pressurization of the Reactor?

A. At ~800 cpm

B. At ~1000 cpm

C. At ~3200 cpm

D. At ~6400 cpm

Answer: D

K/A:

201002 REACTOR MANUAL CONTROL SYSTEM

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

04 Overall reactor power

RO/SRO Rating: 3.6/3.5

Pedigree: New

Objective:
LOI-CLS-LP-307-A, Obj. B6 - GP-02, Approach to Criticality and Pressurization of the Reactor: List the indications that the reactor is critical in accordance with GP-02 (LOCT)

Reference: none

Cog Level: hi

Explanation:
As a rule of thumb, five "doubles" in the neutron count rate will yield criticality.
Initial count rate 200 cpm
1st double = 400 cpm
2nd double = 800 cpm
3rd double = 1600 cpm
4th double = 3200 cpm
5th double = 6400 cpm

Distractor Analysis:

Choice A: Plausible because a common error is to count the initial readings as one of the doubling values with that logic this would be three doublings which is when single notching of control rods is required as the operators approach criticality.

Choice B: Plausible because this value is the current reading times 5.

Choice C: Plausible because a common error is to count the initial readings as one of the doubling values with that logic this would be five doublings.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

NOTE: When performing 'doubling' calculations, start with the values indicated on the SRM s. Example: If the indicated SRM value is 100 cpm then (2 X 100 =200) 200 cpm is the FIRST 'doubling', then 800 cpm is the third 'doubling' recorded in Step 5.2.7.1, and 3200 cpm is the fifth 'doubling' value to be recorded in Step 5.2.7.2.

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NOTE: As a rule of thumb, five "doubles" in the neutron count rate will yield criticality; however, this rule may not always hold true due to initial core conditions and time between control rod withdrawals.

2. 201003 1

Unit Two is in MODE 2 starting up after a refueling outage.
The 2A CRD Pump has tripped and the operator is unable to restart the pump.

The following conditions exist:

Reactor water level	187 inches
Reactor power	Range 8 on all IRM's
Reactor pressure	700 psig
Charging water pressure	700 psig
Reactor temperature	505°F
2B CRD Pump	Out-of-service

Which one of the following identifies when a manual scram is required to be inserted IAW 0AOP-02.0, Control Rod Malfunction/Misposition?

A. If a single control rod scrams.

B. If there are nine or more inoperable rods.

C. If A-05 (3-2) *Rod Drift* alarms due to a single control rod drift.

D. If A-07 (6-1) *CRD Accum Lo Press Hi Level* alarms due to low pressure.

Answer: D

K/A:
201003 CONTROL ROD AND DRIVE MECHANISM
A2 Ability to (a) predict the impacts of the following on the CONTROL ROD AND DRIVE MECHANISM; 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)
09 Low reactor pressure

RO/SRO Rating: 3.2/3.4

Pedigree: Bank, last used on the 2003 NRC exam (modified answer to match current procedure)

Objective:
LOI-CLS-LP-302-B, Obj 4 - Given plant conditions, determine the required supplementary actions IAW 0AOP-02, Control Rod Malfunction/Misposition. (LOCT)

Reference: None

Cog Level: Fundamental Knowledge

Explanation:
Could not write question to both parts of the K/A so wrote the question to the higher cognitive part.
If reactor pressure is less than 950 psig and charging water pressure is less than 940 psig upon the first HCU low pressure alarm immediately insert a manual scram.

Distractor Analysis:

Choice A: Plausible because this is an action in the AOP if more than one control rod scrams.

Choice B: Plausible because the scram is required if any control rod has scrammed and the total inoperable and scrammed rods is greater than nine

Choice C: Plausible because this is an action in the AOP if more than one control rod drifts.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

2. **IF** more than one control rod is drifting,
THEN insert a manual scram **AND enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure.......................... ☐

3. **IF** less than or equal to 25% power,
AND more than one control rod scrams,
THEN insert a manual scram **AND enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure.......................... ☐

CONTROL ROD MALFUNCTION/MISPOSITION	0AOP-02.0
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4.2 Supplementary Actions

NOTE
Section 4.2 Step 1 and Section 4.2 Step 2 only apply if control rod(s) have scrammed .... ☐

1. **IF** the sum of scrammed **AND** INOPERABLE control rods is nine or more,
THEN refer to Technical Specification LCO 3.1.3, Control Rod Operability, for shutdown requirements. .... ☐

CONTROL ROD MALFUNCTION/MISPOSITION	0AOP-02.0
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4.2 Supplementary Actions (continued)

b. **IF** reactor pressure is less than 950 psig,
AND CRD charging pressure can **NOT** be restored to greater than or equal to 940 psig with either CRD pump,
THEN upon receipt of the first HCU low pressure alarm (A-07 6-1, confirmed by amber light on Full Core Display),
immediately insert a manual scram **AND enter** 1EOP-01-RSP(2EOP-01-RSP), Reactor Scram Procedure............... ☐

3. 201006 1

Unit One is operating at 32% power when one of the four Main Steam Line Flow Transmitter inputs to the Feedwater Level Control System has failed downscale.

Which one of the following identifies the effect this will have, if any, on the RWM?

The RWM will:

A. display BYPASSED.

B. provide alarms ONLY.

C. provide alarms and enforce rod blocks.

D. NOT provide alarms or enforce rod blocks.

Answer: B

K/A:

201006 ROD WORTH MINIMIZER SYSTEM (RWM)

K6 Knowledge of the effect that a loss or malfunction of the following will have on the ROD WORTH MINIMIZER SYSTEM (RWM): (CFR: 41.7 / 45.7)

05 Steam flow input

RO/SRO Rating: 2.7/2.7

Pedigree: New

Objective:

LOI-CLS-LP-07.1, Obj 3 - Describe the operation of the RWM above and below the Low Power Setpoint (LPSP) and the Low Power Alarm Point (LPAP), including the setpoints and where the input signal originates.

Reference: None

Cog Level: Fundamental Knowledge

Explanation:

Transition zone is set when steam flow is between 19.1% and 27.8%. With power initially at 32%, a single steam flow indicator failing will result in a total steam flow signal lowering to 24%. At 24% steam flow alarms are active but rod blocks are not enforced.

Distractor Analysis:

Choice A: Plausible because the RWM may need to be bypassed if power was dropped to less than 19.1%.

Choice B: Correct Answer, see explanation

Choice C: Plausible because if power was such that the failure caused power to drop below 19.1% this would be correct.

Choice D: Plausible because with power at 32% alarms and rod blocks are not provided.

SRO Basis: N/A

4. 203000 1

The BOP operator is aligning RHR Loop B to transfer water from the Suppression Pool to the Auxiliary Surge Tank.

Which one of the following identifies how far the suppression pool level is expected to drop if 3,100 gallons is transferred IAW 1OP-17, Residual Heat Removal System Operating Procedure?

A. ~½ inch

B. ~1½ inch

C. ~4 inches

D. ~5 inches

Answer: A

K/A:

203000 RHR/LPCI: INJECTION MODE

A1 Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE controls including: (CFR: 41.5 / 45.5)

05 Suppression pool level

RO/SRO Rating: 3.8/3.7

Pedigree: New

Objective:

AOI-CLS-LP-006, Obj 2 - Draw/Discuss the flow path/circuit path associated with the Liquid Radwaste System to include the major components, normal, abnormal, and secondary flow paths, and interrelationships with other systems. x. Auxiliary Surge Tank

Reference: none

Cog Level: hi

Explanation:

One inch in the suppression pool corresponds to approximately 6,200 gallons in the normal operating band. The Aux Surge Tank is 7460 gallons/foot or 622 gallons/inch.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because if the math is performed incorrectly this value can be obtained.

Choice C: Plausible because this is the normal level band range (-27 inches to -31 inches) for the suppression pool.

Choice D: Plausible because the aux surge tank changes 622 gal/inch which would be ~5 inch change.

SRO Basis: N/A

NOTE: One inch in the suppression pool corresponds to approximately 6,200 gallons in the normal operating band.

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5. 203000 2

Unit Two was operating at rated power when a LOCA occurred. ADS has automatically initiated and reactor pressure is lowering.

Which one of the following identifies the highest reactor pressure that will allow RHR injection flow to be seen on E11-FI-R603B, RHR System B Flow?

A. ~400 psig.

B. ~300 psig.

C. ~200 psig.

D. ~100 psig.

Answer: C

K/A:
203000 RHR/LPCI: INJECTION MODE
A3 Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE including: (CFR: 41.7 / 45.7)
03 Pump discharge pressure

RO/SRO Rating: 3.7/3.6

Pedigree: NRC 2008 exam question

Objective:
LOI-CLS-LP-017, Obj 7 - Given plant conditions, determine if the RHR System should automatically initiate in the LPCI mode.

Reference: None

Cog Level: Fundamental knowledge

Explanation:
As reactor pressure continues to decrease, the discharge of the RHR Pumps should overcome reactor pressure at approximately 200 psig, allowing the flowpath to continue into the Reactor Recirculation System discharge lines.

Distractor Analysis:

Choice A: Plausible because this is the approx. the reactor pressure that the discharge valve gets an open signal at.

Choice B: Plausible because this is the reactor pressure that Core Spray will inject at.

Choice C: Correct Answer, see explanation

Choice D: Plausible because this is the pressure needed to satisfy the pump running logic for ADS.

SRO Basis: N/A

With the RHR and CS Pumps running, indicated pump discharge pressure on CS should increase to approximately 305 psig and RHR should be about 200 psig. As reactor pressure decreases to approximately 410 psig, the LPCI Inboard Injection Valve, E11-F015A(B), should automatically open. As reactor pressure continues to decrease, the discharge of the RHR Pumps should overcome reactor pressure at approximately 200 psig, allowing the flowpath to continue from the RHR Pumps' discharge check valve directly into the Reactor Vessel through the normally open LPCI Outboard Injection Valve, E11-F017A(B), the LPCI Inboard Injection Valve, E11-F015A(B), the LPCI Injection Line Check Valve, E11-F050A(B), the locked open LPCI Manual Injection Valve, E11-F060A(B), and into the Reactor Recirculation System discharge lines. Once reactor pressure is reduced to approximately 20 psig, RHR flow should reach approximately 17,000 gpm per operating loop with two pumps.

6. 205000 1

Unit Two is in day 4 of a refueling outage with RHR Loop 2B operating in Shutdown Cooling IAW 2OP-17, Residual Heat Removal System Operating Procedure.

Which one of the following completes the statements below?

The lowest reactor pressure that will cause a Group 8 isolation is ___(1)___ psig.

The Group 8 isolation pressure signal ___(2)___ cause E11-F015B, Inboard Injection Vlv to auto close?

A. (1) ~135
   (2) will

B. (1) ~135
   (2) will NOT

C. (1) ~200
   (2) will

D. (1) ~200
   (2) will NOT

Answer: B

K/A:

205000 SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE)

K4 Knowledge of SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

05 Reactor cooldown rate

RO/SRO Rating: 3.6/3.7

Pedigree: New

Objective:

LOI-CLS-LP-017, Obj. 14 - Given plant conditions, determine if a Shutdown Cooling isolation should have occurred.

Reference: None

Cog Level: High

Explanation:

Reactor pressure rising is an indication of a heat up rate in the reactor. If pressure rises to 130.8 psig then an isolation signal will be generated for the SDC suction isolation valves (F008 & F009). If level is low then the F015 also would close. 200 psig is the shutoff head of the RHR pumps.

Distractor Analysis:

Choice A: Plausible because 135 psig is correct and if level was also low then the F015 would close.

Choice B: Correct Answer, see explanation

Choice C: Plausible because 200 psig is the shutoff head for the RHR pumps and if level was also low then the F015 would close.

Choice D: Plausible because 200 psig is the shutoff head for the RHR pumps and the F015 valve will not close on a pressure signal.

SRO Basis: N/A

Before the isolation valves (E11-F008 and F009) can be opened, the following permissives must be satisfied:

- Reactor vessel water level is greater than low level 1 as proven by level trip units B21-LTM-N017A-1 and B-1 for valve E11-F009 and B21-LTM-N017C-1 and D-1 for valve E11-F008.
- Reactor pressure is less than 130.8 psig (which is about 350°F) as proven by pressure switches B32-PS-N018A-1 and N018B SW #2.
- Keylocked test switches (A71-S22A or B for valve E11-F009 and A71-S22C or D for valve E11-F008) are in NORMAL position.
- PCIS reset push buttons (A71-S32 for valve E11-F009 and A71-S33 for valve E11-F008) have been depressed.

The occurrence of any of the above will not affect the position of the shutdown cooling suction valves (E11-F006A through D); i.e., they will remain open if initially open or remain closed if initially closed.

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If the RHR shutdown cooling isolation valves E11-F008 and F009 are open and reactor vessel water level drops below reactor vessel low level 1, the Group 8 Isolation Valves E11-F008, F009, and the F015 A and B valves will automatically close and the operating pump(s) will trip.

The E11-F008 and F009 will also close and the associated RHR pump(s) trip if reactor vessel pressure increases over 135 psig.

7. 206000 1

During accident condition on Unit Two the following plant conditions exist:

RPV water level	-30 inches
Reactor power	4%
Suppr pool temp	142°F
Suppr pool level	-24 inches

HPCI operation is required.

Which one of the following identifies:
(1) the preferred suction source for HPCI and
(2) the reason that suction source is preferred?

A. (1) Suppression pool.
(2) To prevent continued rise in suppression pool level.

B. (1) Suppression pool.
(2) To provide a warmer source of injection to the reactor.

C. (1) CST.
(2) To prevent damage to the HPCI pump due to cavitation.

D. (1) CST.
(2) To prevent overheating of HPCI lubricating and control oil.

Answer: D

K/A:
206000 HIGH PRESSURE COOLANT INJECTION SYSTEM
A2 Ability to (a) predict the impacts of the following on the HIGH PRESSURE COOLANT INJECTION 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)
08 High suppression pool temperature: BWR-2,3,4

RO/SRO Rating: 3.9/4.2

Pedigree: Bank question

Objective:
LOI-CLS-LP-019, Obj 3 - Given plant conditions, predict how the HPCI System will respond to the following events: y. High Suppression Pool temperature

Reference: None

Cog Level: Fundamental Knowledge

Explanation:
The lube oil and control oil for both HPCI and RCIC are cooled by the water being pumped. Very high lube oil temperatures can result in loss of lubricating qualities in the oil and thus cause damage to the bearings. Suction for HPCI and RCIC is aligned to the Condensate Storage Tank (CST) if it is available. The HPCI automatic suction transfer logic can be defeated to allow this lineup if necessary provided suppression pool temperature is approaching 140°F.

Distractor Analysis:

Choice A: Plausible because at high level in the torus it does transfer and if the temperature was <140°F this would be correct.

Choice B: Plausible because at high level in the torus it does transfer and if the temperature was <140°F this would be correct and the suppression pool water is warmer than the CST.

Choice C: Plausible because transferring to the CST is correct and level at -24 inches might be considered low.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

CAUTION

OPERATION OF HPCI OR RCIC WITH SUCTION TEMPERATURES ABOVE 140°F MAY RESULT IN EQUIPMENT DAMAGE

The lube oil and control oil for both HPCI and RCIC are cooled by the water being pumped. Very high lube oil temperatures can result in loss of lubricating qualities in the oil and thus cause damage to the bearings.

2. Defeat of the HPCI High Suppression Pool Level Suction Transfer

   This logic should be defeated if plant conditions will result in the suppression pool temperature approaching 140°F when HPCI operation may be needed. The HPCI system is designed to operate for a short time above 140°F.

8. 206000 2

Which one of the following identifies the Unit Two HPCI turbine speed control power supply?

A. 125 VDC Panel 3A

B. 125 VDC Panel 3B

C. 125 VDC Panel 4A

D. 125 VDC Panel 4B

Answer: C

K/A:
206000 HIGH PRESSURE COOLANT INJECTION SYSTEM
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
04 Turbine control circuits: BWR-2,3,4

RO/SRO Rating: 2.5/2.7

Pedigree: New

Objective:
LOI-CLS-LP-019, Obj 14c - State the power supplies (bus and voltage) for the following HPCI System components: HPCI Flow Controller

Reference: None

Cog Level: Memory

Explanation:
The turbine speed control, both the 24 VDC power supply which powers the flow controller and the 52.5 VDC power supply which powers the instrumentation, is powered from Distribution Panel 3(4)A.

Distractor Analysis:

Choice A: Plausible because this panel supplies Unit One HPCI.

Choice B: Plausible because one of the isolation logics on Unit One is powered from here.

Choice C: Correct Answer, see explanation

Choice D: Plausible because one of the isolation logics is powered from here.

SRO Basis: N/A

TABLE 19-11

HPCI System Power Supplies

COMPONENT	NUMBER	POWER SOURCE	COMPT.
HPCI Barometric Condenser Vacuum Pump	E41-C002 Vac-Pmp	1(2)XDA	B13
HPCI Flow Controller	E41-FIC-R600	125 VDC Bus A	Panel 3A Ckt. #2
HPCI Turbine Test Circuitry		Emergency 120 VAC	Panel 1(2)A, Ckt. #11
HPCI Turbine Speed Control (RGSC, EG-M)		125 VDC Bus A	Panel 3(4)A, Ckt. #2
HPCI Relay Logic, Trip Circuitry & Isolation Logic Bus A		125 VDC Bus A	Panel 3(4)A, Ckt. #13
Isolation Logic Bus B		125 VDC Bus B	Panel 3(4)B Ckt. #3
Leak Detection Logic Div. I		1(2) CA	Panel 31A(32A) Ckt. #1
Leak Detection Logic Div. II		1(2) CB	Panel 31B(32B) Ckt. #1

9. 209001 1

Unit Two is operating at rated power when the following alarm is received:

A-01 (2-10) *Core Spray Loop A Sys Press Low*

Which one of the following identifies the impact of this condition on the Core Spray System?

A. Core Spray Pump A may cause piping damage, if started.

B. Core Spray Pump A is incapable of producing an ADS Logic permissive signal, if started.

C. E21-F005A, Inboard Inject Valve, will immediately open if the Core Spray Initiation Logic is actuated.

D. E21-F004A, Outboard Inject Valve, can be opened while E21-F005A, Inboard Inject Valve, is open.

Answer: A

K/A:
209001 LOW PRESSURE CORE SPRAY SYSTEM
K5 Knowledge of the operational implications of the following concepts as they apply to LOW PRESSURE CORE SPRAY SYSTEM: (CFR: 41.5 / 45.3)
05 System venting

RO/SRO Rating: 2.5/2.5

Pedigree: New

Objective:
LOI-CLS-LP-018, Obj 6 - Describe how "water hammer" is minimized in the Core Spray System.

Reference: None

Cog Level: Fundamental knowledge

Explanation:
Core Spray Discharge Pressure <10.8 psig causes the listed annunciator, indicating that piping may no longer be properly filled and vented. Voided piping may be damaged by starting the Core Spray Pump.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because this would be true if Core Spray or RHR Pump Running (115 psig) were failed low

Choice C: Plausible because this would be true if Reactor Pressure below 410 psig failed low.

Choice D: Plausible because this would be true if Reactor Pressure below 410 psig and the valves were reversed, 5 can be opened if the 4 is opened first.

SRO Basis: N/A

CORE SPRAY LOOP A SYS PRESS LOW

AUTO ACTIONS

NONE

CAUSE

1. Keepfill Station Pressure Control Valve, E21-PCV-F026A, failure or valve lineup incorrect
2. Discharge header not charged or leaking
3. Circuit malfunction

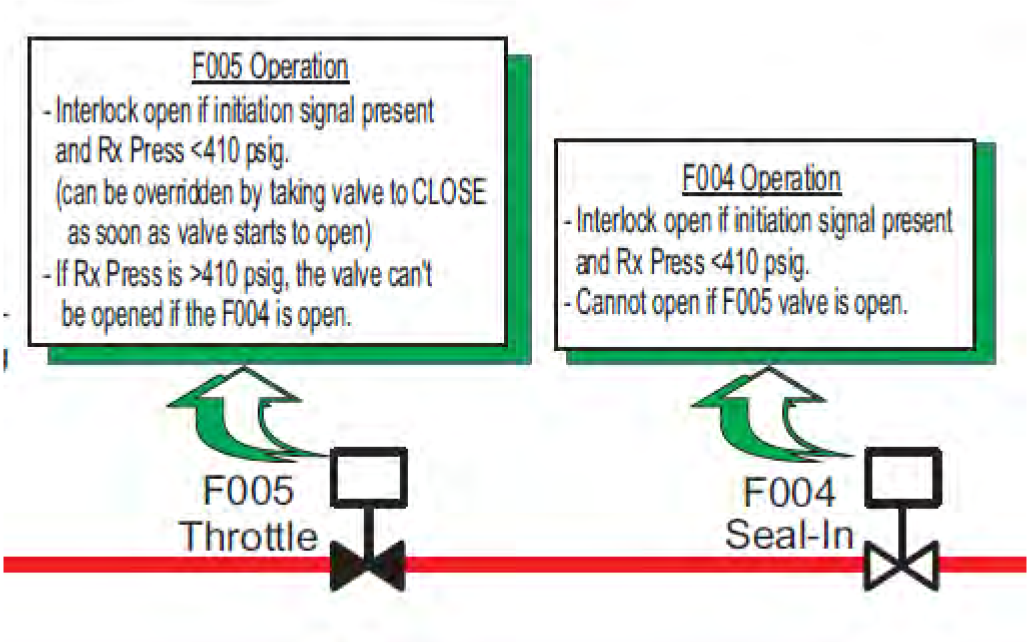
POSSIBLE PLANT EFFECTS

1. Actuation of Core Spray when discharge header is not charged may result in severe water hammer and pipe damage.
2. If discharge header cannot be kept pressurized, a technical specification LCO may result.

CORE SPRAY OR RHR PUMPS RUNNING

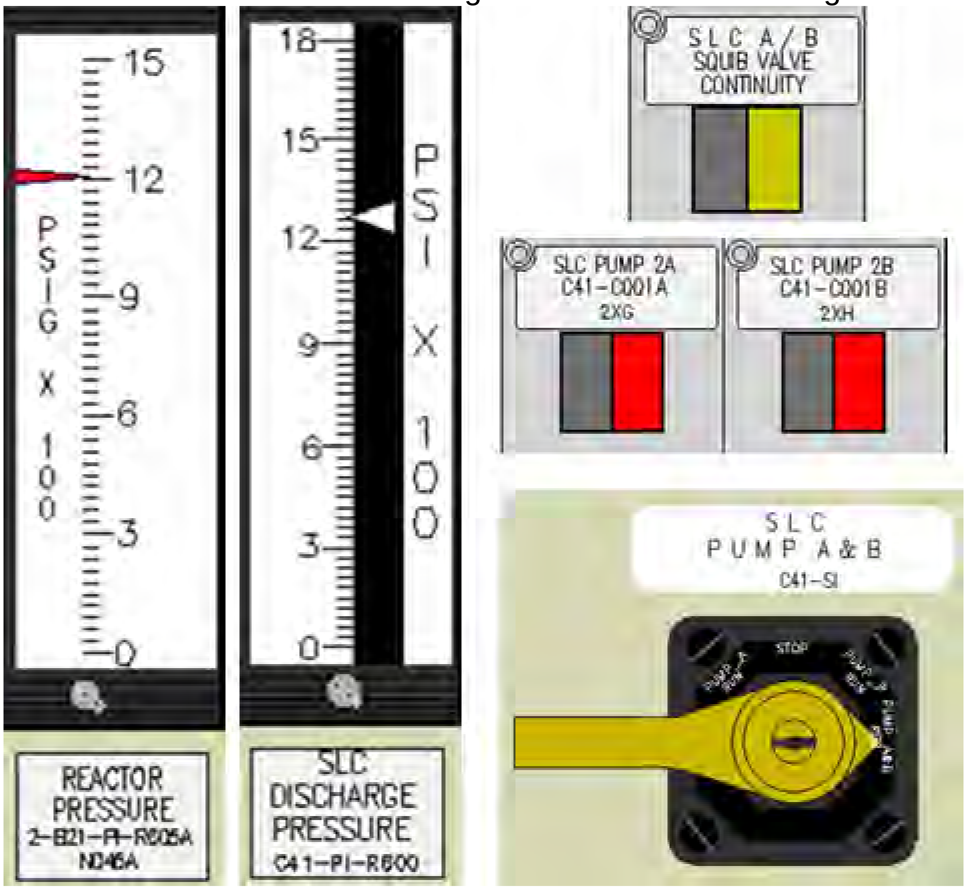
DEVICE/SETPOINTS

Pressure Switch E21-PS-N008A, B	114 psig, with -1 psig head correction
Pressure Switch E21-PS-N009A, B	114 psig, with -1 psig head correction
Pressure Switch E11-PS-N016A, B, C, D	117 psig, with 2 psig head correction
Pressure Switch E11-PS-N020A, B, C, D	117 psig, with 2 psig head correction



10. 211000 1

The OATC observes the following indications after initiating SLC during an ATWS.



Which one of the following completes the statements below?

Squib valve ___(1)___ has failed to fire.

IAW 2OP-05, Standby Liquid System Operating Procedure, the OATC is required to ___(2)___.

A. (1) A
   (2) place the CS-S1, SLC Pump A & B, in the PUMP A RUN position

B. (1) A
   (2) leave the CS-S1, SLC Pump A & B, in the PUMP A/B RUN position

C. (1) B
   (2) place the CS-S1, SLC Pump A & B, in the PUMP A RUN position

D. (1) B
   (2) leave the CS-S1, SLC Pump A & B, in the PUMP A/B RUN position

Answer: C

K/A:

211000 STANDBY LIQUID CONTROL SYSTEM

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

02 SBLC control switch

RO/SRO Rating: 4.2/4.2

Pedigree: New

Objective:
LOI-CLS-LP-005, Obj 13 **-** Predict the effect of the following on the Standby Liquid Control System, and based on those predictions use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: a. Failure of one or both squib valves to fire. **(LOCT)**

Reference: None

Cog Level: Hi

Explanation:
The SLC squib valve continuity lights are normally lit and go out when fired on SLC initiation. Per OP-05, if one squib valve fails to fire, two pump SLC operation may still continue provided reactor pressure is below 1184 psig, which it is not.

Distractor Analysis:

Choice A: Plausible because the student may think that the light is illuminated when the squib valve fires and securing 1 pump is correct.

Choice B: Plausible because the student may think that the light is illuminated when the squib valve fires and if reactor pressure was lower this would be correct.

Choice C: Correct Answer, see explanation

Choice D: Plausible because the B squib did not fire and if reactor pressure was lower this would be correct.

SRO Basis: N/A

NOTE:	The SLC pump discharge relief valve should **NOT** actuate with two pumps operating and only one squib valve open unless reactor pressure exceeds 1184 psig, which is possible during an ATWS even with 10 SRVs open.

2. **IF** *SLC A SQUIB VALVE CONTINUITY* **OR** *SLC B SQUIB VALVE CONTINUITY* indicating light on Panel P603 remains on **AND** reactor pressure is greater than or equal to 1184 psig, **THEN PERFORM** the following:

   a. **PLACE** *SLC PUMP A & B* Control Switch, *C41-CS-S1*, to the *SLC PUMP A* **OR** *SLC PUMP B* position. ☐

   b. **ENSURE** the selected SLC pump red indicating light on. ☐

11. 212000 1

Which one of the following identifies which RPS MG Set and EPA breakers that trip on a loss of 480 VAC Substation E7?

RPS MG Set ___(1)___ EPA breakers ___(2)___.

A. (1) A
(2) 1 & 2 ONLY

B. (1) B
(2) 3 & 4 ONLY

C. (1) A
(2) 1 & 2 and alternate source EPA breakers 5 & 6

D. (1) B
(2) 3 & 4 and alternate source EPA breakers 5 & 6

Answer: C

K/A:
212000 REACTOR PROTECTION SYSTEM
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
01 RPS motor-generator sets

RO/SRO Rating: 3.2/3.3

Pedigree: Systems bank

Objective:
CLS-LP-03 Obj 18a - State the power supplies for the following: RPS MG Set A

Pedigree: 10-1 NRC Exam

Reference: None

Cog Level: Memory

Explanation:
Power for the Motor Generator Sets is tapped off two phases of the normal 480 VAC MC 1CA/1CB (2CA/2CB) power supply for the motor through a stepdown transformer (480V to 120V) from E5/E6 (E7/E8). Selectable reserve power to the Bus is provided from 120 VAC 1E5(2E7) or 1E6(2E8), and is normally selected to Division I. In the event that either RPS M-G Set fails to operate, the alternate power source must be manually selected.
Two EPAs in series are installed downstream of the generator output breaker for each Motor Generator Set and the alternate power supply for the RPS buses. Bus A is protected by EPA-1 and -2; Bus B by EPA-3 and -4. Alternate power is protected by EPA-5 and -6

Distractor Analysis:

Choice A: Plausible because A MG set is lost along with EPA breakers 1 & 2, but these are not the only EPA breakers to trip.

Choice B: Plausible if the examinee picks the wrong power supply and EPA breakers 3 & 4 are powered from RPS MG Set B.

Choice C: Correct Answer, see explanation

Choice D: Plausible if the examinee picks the wrong power supply and EPA breakers 3 & 4 are powered from RPS MG Set B.

SRO Basis: N/A

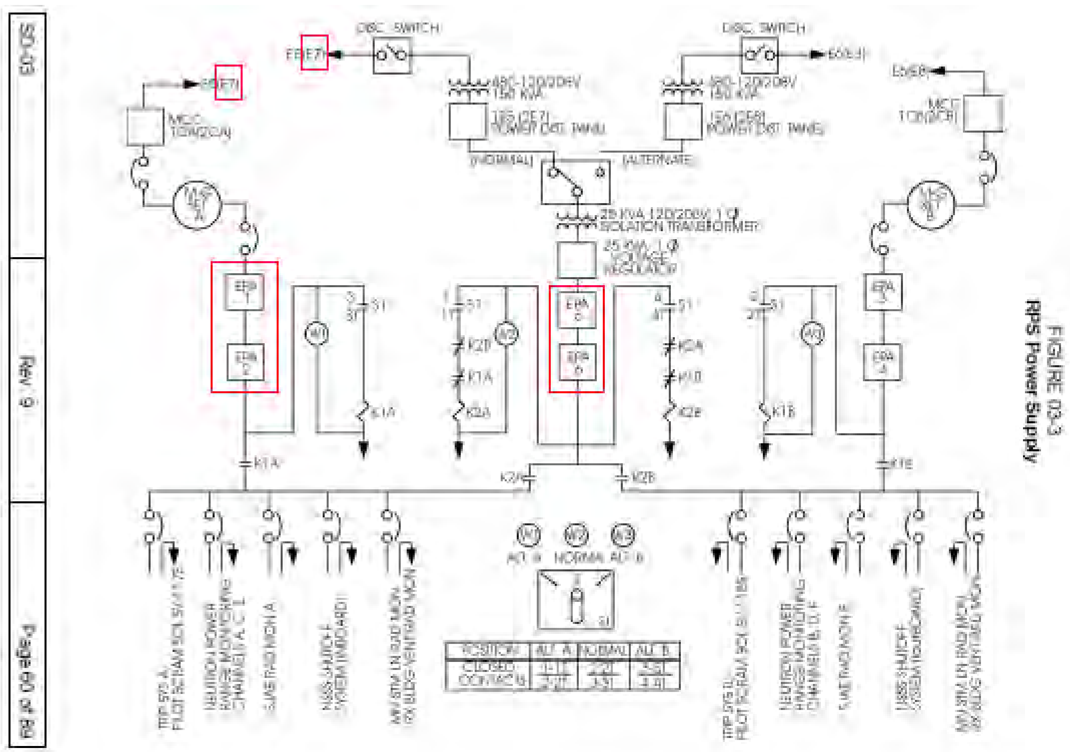
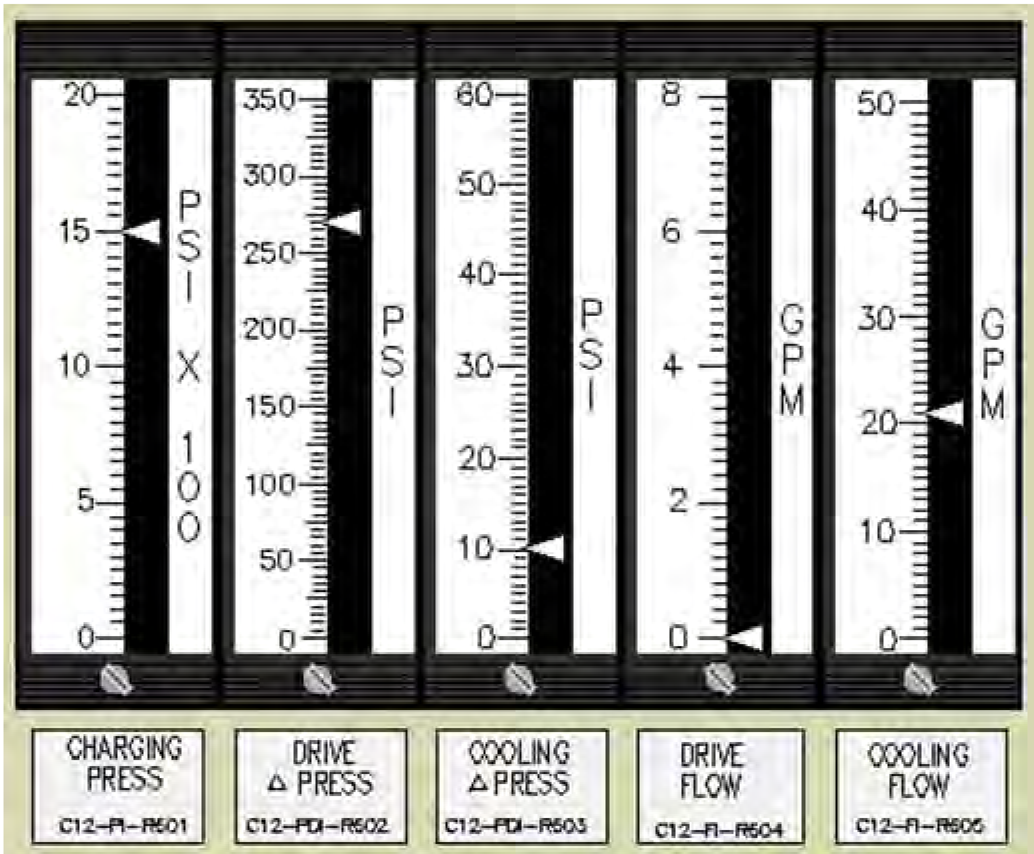


FIGURE 03-3 RPS Power Supply

12. 214000 1

Unit Two is operating at rated power when A-05 (1-2) *CRD Hyd Temp High* annunciates and the OATC observes the following CRD indications.



Which one of the following completes the statements below?

A-05 (1-2) *CRD Hyd Temp High* setpoint is ___(1)___°F.

The APP will provide guidance to adjust C11-FC-R600, CRD Flow Controller, in the ___(2)___ direction.

A. (1) 350
   (2) open

B. (1) 350
   (2) closed

C. (1) 340
   (2) open

D. (1) 340
   (2) closed

Answer: C

K/A:
214000 ROD POSITION INFORMATION SYSTEM
A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
03 Control rod drive temperature

RO/SRO Rating: 2.8/2.7

Pedigree: New

Objective:
LOI-CLS-LP-008, Obj 8 - Given plant conditions, predict the effect that a loss or malfunction of the following will have on the CRDH System: m. Cooling Water Flow

Reference: None

Cog Level: High

Explanation:
The alarm setpoint for the annunciator is 340°F. There is an action it exceeds 350°F notifying an engineer to perform a PT). raising flow would require the controller to be opened. The Drive water control valve (F003) could also be closed to raise the flow.

Distractor Analysis:

Choice A: Plausible because 250°F is addressed in the APP and open is correct.

Choice B: Plausible because 250°F is addressed in the APP and if the question asked for the operation of the drive water control valve (F003) this would be correct.

Choice C: Correct Answer, see explanation

Choice D: Plausible because 340°F is correct and if the question asked for the operation of the drive water control valve (F003) this would be correct.

SRO Basis: N/A

2. If cooling water flow is low, raise the CRDHS flow rate per OP-08, CRDHS.

ACTIONS (Continued)

4. Monitor scram discharge valve for leakage by feeling exhaust piping on HCU, or using a pyrometer.
   a) If scram discharge valve is determined to be leaking-by, contact Reactor Engineering to provide guidance for insertion of control rod in case control rod insertion is deemed necessary to prevent rod drift.
5. If indicated temperature cannot be reduced below 250°F, by allowing a single notch insert and withdraw, initiate a WR to investigate.
6. If the temperature cannot be maintained below the alarm setpoint, disable the alarm for the affected CRD on CRD/Reactor Vessel Temperature Recorder, C12-TR-R018, to allow detecting future hot drives.
7. If the CRD temperature exceeds 350°F for more than one week, then notify Engineering to schedule 0PT-14.2.1 for scram time testing.
8. Notify System Engineer if the drive temperature is above 350°F to determine effects on CRD scram time.

DEVICE/SETPOINTS

CRD/Reactor Vessel Temperature Recorder C12-TR-R018	340°F

13. 215003 1

Which one of the following completes the statement below?

The Intermediate Range Monitor (IRM) detectors may be positioned full in ___(1)___, full out ___(2)___, or any intermediate position.

A. (1) 18 inches above the core centerline
(2) bottom of the core

B. (1) 18 inches above the core centerline
(2) 24 inches below the core

C. (1) at the core centerline
(2) bottom of the core

D. (1) at the core centerline
(2) 24 inches below the core

Answer: B

K/A:
215003 INTERMEDIATE RANGE MONITOR (IRM) SYSTEM
K1 Knowledge of the physical connections and/or cause effect relationships between INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
07 Reactor vessel

RO/SRO Rating: 3.0/3.0

Pedigree: Bank, from Nine Mile Point 2008 NRC exam

Objective:
LOI-CLS-LP-009-A, Obj. 2 - State the purpose and/or function of the following components pertaining to the SRM and IRM systems as applicable: b. Drive Unit

Reference: None

Cog Level: fundamental knowledge

Explanation:
Both the SRM and IRM detectors are positioned within the reactor core by means of detector insert and retract mechanism which is controlled from the reactor control benchboard. The detectors may be positioned full in (18 inches above the core centerline), full out (24 inches below the core), or any intermediate position

Distractor Analysis:

Choice A: Plausible because bottom of active fuel is also another common reference point and a plausible misconception

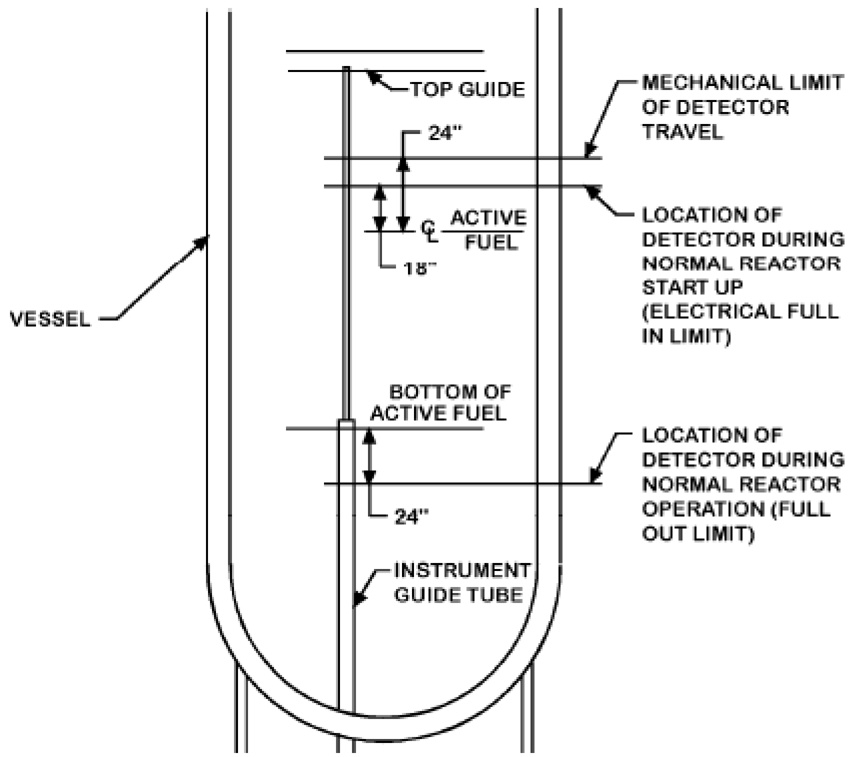
Choice B: Correct Answer, see explanationPlausible because

Choice C: Plausible because core centerline is a common reference point and can be a plausible misconception, and bottom of active fuel is also another common reference point and a plausible misconception

Choice D: Plausible because core centerline is a common reference point and can be a plausible misconception.

SRO Basis: N/A

Both the SRM and IRM detectors are positioned within the reactor core by means of detector insert and retract mechanism which is controlled from the reactor control benchboard. The detectors may be positioned full in (18 inches above the core centerline), full out (24 inches below the core)(see Figure 09.1-10), or any intermediate position.



14. 215004 1

The reactor has just been declared critical during a reactor startup IAW 0GP-02, Approach To Criticality and Pressurization of the Reactor, with SRM channel A bypassed. A-05 (2-2) *Rod Out Block* and A-05 (2-3) *SRM Upscale/Inop* are in alarm.

As the operator attempts to withdraw SRM B detector is stuck and will not retract from the full in position.

Which one of the following completes the statement below?

Rods cannot be withdrawn until the IRMs are ranged to Range _________ or above.

A. 2

B. 3

C. 7

D. 8

Answer: D

K/A:
215004 SOURCE RANGE MONITOR (SRM) SYSTEM
K3 Knowledge of the effect that a loss or malfunction of the SOURCE RANGE MONITOR (SRM) SYSTEM will have on following: (CFR: 41.7 / 45.4)
02 Reactor manual control

RO/SRO Rating: 3.4/3.4

Pedigree: Bank, last used on the 2008 NRC exam

Objective:
LOI-CLS-LP-009.1, Obj 10 - Given plant conditions, predict the response of the SRM/IRM system to a malfunction/failure of the following systems/components: e. Detector Drive motor

Reference: None

Cog Level: Fundamental Knowledge

Explanation:
SRM Channel B detector is stuck. Since the reactor is critical power will continue to increase until the heating range is reached (range 7 to 8 of IRMs). The SRM rod block is at 2 X 10^5 counts/second. Only one SRM channel at a time can be bypassed (unlike IRMs) and SRM A is already bypassed due to a low voltage supply which is an inop trip. When SRM B goes above the rod block setpoint, the alarm procedure directs the operator to bypass SRM B, but bypassing SRM B would require unbypassing A and would not get rid of the rod block. Some SRM rod blocks (downscale and detector retract) are bypassed with IRMs on range 3 or above, but the upscale high and inop rod blocks are not bypassed until IRMs are on range 8 or above. Some of the wording in different documents states bypass when greater than range 2 or 7 for their respective blocks.

Distractor Analysis:

Choice A: Plausible since the setpoint for downscale and detector retract is greater than range 2.

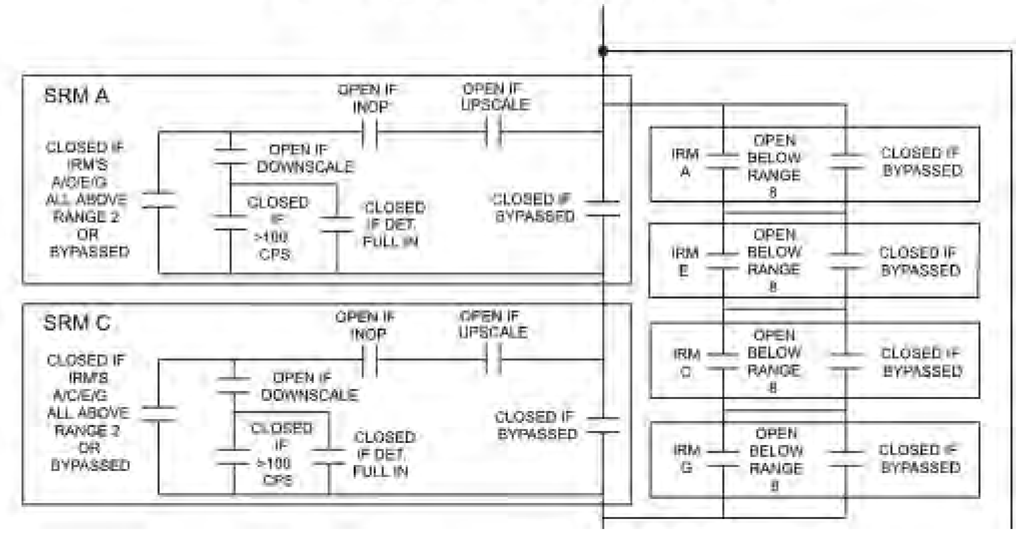
Choice B: Plausible since the setpoint downscale and detector retract is range 3 or above.

Choice C: Plausible since the setpoint for the Upscale is greater than range 7.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

FIGURE 09.1- 12
SRM/IRM Rod Block Circuitry (Channel A)



INTERLOCKS

Trip	Setpoint	Trip Result	Bypassed	Bases
SRM Upscale Trip	5 X 10^5 CPS	Alarm & Scram (when shorting links removed)	Used only during refueling and low power physics testing.	Indicates excessive flux levels and detects changes in reactivity when criticality is possible.
SRM Upscale Alarm	2X 10^5 CPS	Alarm/Rod Block	1. IRMs $\geq$ Range 8. 2. Mode Switch in Run	Generates interlock signals to block control rod withdrawal if count rate exceeds a preset value
SRM INOP	1. High Voltage Low 2. Module Unplugged 3. Function Switch Not in Operate	Alarm/Rod Block	1. IRMs $\geq$ Range 8. 2. Mode Switch in Run	Block generated because the channel may be unreliable.
SRM Downscale	3 CPS TS 5 CPS nominal value	Alarm/Rod Block	1. IRMs $\geq$ Range 3 2. Mode Switch in Run	Malfunction or insufficient source neutron level for safe startup.
SRM Retract Permit	125 CPS (Nominal) 101-150 CPS (Allowable range)	Alarm/Rod Block	1. SRMs Full In 2.IRMs $\geq$ Range 3 3. Mode Switch in Run	Ensures detector is properly positioned for monitoring of control rod movement during startup conditions
SRM Period	50 seconds	Alarm		

SRM UPSCALE/INOP

AUTO ACTIONS

1. Rod withdrawal block (bypassed when all IRM channels are above Range 7 or the reactor mode switch is in RUN).
2. Computer printout.

15. 215004 2

A plant startup is in progress. The OATC was withdrawing SRMs when a control rod block occurred. The following nuclear instrument indications are noted:

SRM	Counts	Position	IRM	Counts	Range
A	95	Mid Position	A	25/125	3
B	190	Mid Position	B	65/125	2
C	$6x10^4$	Full In	C	35/125	3
D	155	Mid Position	D	15/125	3
			E	12/125	2
			F	55/125	3
			G	30/125	2
			H	25/125	3

Which one of the following actions will clear the control rod block?

A. Inserting SRM A.

B. Withdrawing SRM C.

C. Ranging IRM G ONLY to range 3.

D. Ranging IRM B and G to range 3.

Answer: A

K/A:
215004 SOURCE RANGE MONITOR (SRM) SYSTEM
K5 Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM : (CFR: 41.5 / 45.3)
03 Changing detector position

RO/SRO Rating: 2.8/2.8

Pedigree: New

Objective:
CLS-LP-09.1 Obj. 9 - Describe the insertion/withdrawal of the SRM detectors, including the following:
a. Reason for maintaining counts between 125 and $2x10^5$

Reference: None

Cog Level: High

Explanation:
To clear the rod block SRM must be above 125 counts or the divisional IRMs must be $\geq$ range 3, or the mode switch in RUN. Inserting SRM A to get counts around 125 will cause the rod block to clear.

Distractor Analysis:

Choice A: Correct Answer, see explanation

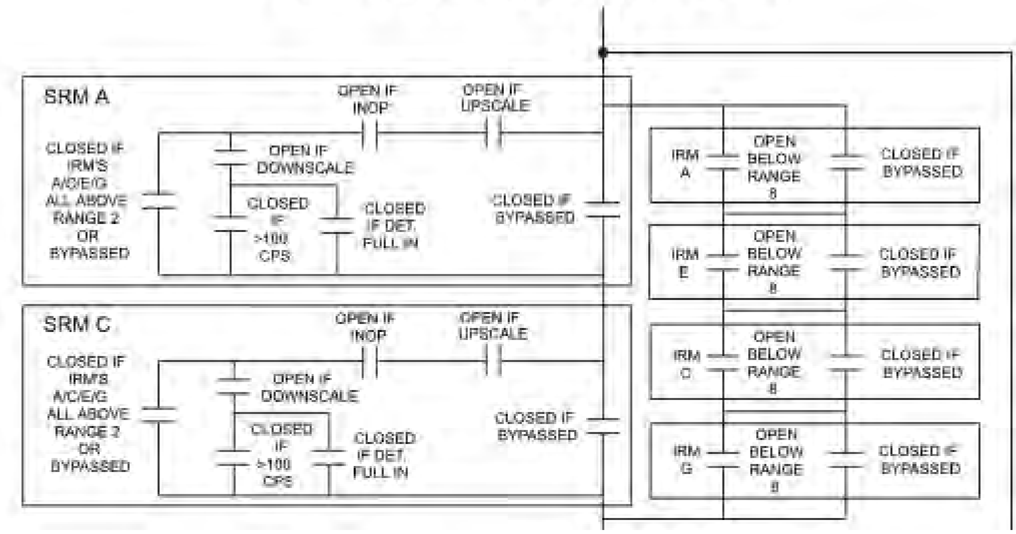
Choice B: Plausible because SRM A does need to be withdrawn and C is above the old setpoint for the upscale alarm. (recent change, old setpoint was $5x10^4$).

Choice C: Plausible because IRM G is a Div I IRM below range 3, but IRM G is also less than range 3. If all Div II IRMs are above range 3 then the rod block from SRM Retract Permissive in would be bypassed

Choice D: Plausible because IRM B & G are below range 3. This would still leave IRM E on Div I below range 3. If all Div I IRMs are above range 3 then a rod block from SRM Retract Permissive would be bypassed on Div I.

SRO Basis: N/A

FIGURE 09.1- 12
SRM/IRM Rod Block Circuitry (Channel A)



16. 215005 1

Unit Two is operating at rated power with the following conditions:

A-05 (2-2) *Rod Out Block*	In alarm
A-05 (4-8) *OPRM Trip Enabled*	NOT in alarm
A-06 (2-8) A*PRM Upscale*	NOT in alarm
A-06 (5-7) *Flow Ref Off Normal*	In alarm

Which one of the following completes the statements below?

A total recirc flow channel has failed ___(1)___.

IAW APP A-06 (5-7) *Flow Ref off Normal,* the OATC will ___(2)___ the affected APRM.

A. (1) downscale
(2) bypass

B. (1) downscale
(2) place the mode switch in INOP for

C. (1) upscale
(2) bypass

D. (1) upscale
(2) place the mode switch in INOP for

Answer: C

K/A:
215005 AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM
A2 Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR 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)
06 Recirculation flow channels upscale

RO/SRO Rating: 3.4/3.5

Pedigree: New

Objective:
LOI-CLS-LP-009.6, Obj 6e - Given plant conditions, predict the response of the PRNMS to a malfunction/failure of the following systems/components: Recirc Flow Module (LOCT)

Reference: None

Cog Level: High

Explanation:
These are the only two alarms that annunciate for a channel failing upscale. If it were to fail downscale APRM upscale, APRM upscale Trip/Inop, and OPRM enabled alarms would annunciate. The APP has the operator bypass the APRM. The action to place the mode switch in INOP is from OI-18 to place the channel in a tripped condition.

Distractor Analysis:

Choice A: Plausible because these alarms do annunciate on failing downscale and bypassing the APRM is correct.

Choice B: Plausible because these alarms do annunciate on failing downscale and this is the action for placing the channel in a tripped condition.

Choice C: Correct Answer, see explanation

Choice D: Plausible because upscale is correct (only these two alarms would be received) and this is the action for placing the channel in a tripped condition.

SRO Basis: N/A

FLOW REF OFF NORMAL

AUTO ACTIONS

1. Rod Withdrawal Block if cause of the alarm is from a Total Recirculation Flow upscale condition from one of the APRMs.

CAUSE

1. Total Recirc Flow upscale.
2. Total Recirc Flow comparator alarm from RBM.
3. Flow transmitter failure for Total Recirc Flow instrumentation.

OBSERVATIONS

1. If alarm due to Comparator, then FLOW COMPARE in inverse video shown on RBM ODA/NUMAC header.
2. ROD OUT BLOCK (A-05 2-2) alarm, if Total Recirc Flow upscale condition.
3. FLOW (%) greater than 110% indicated on APRM ODA/NUMAC BARGRAPH.
4. If flow transmitter fails downscale, APRM UPSCALE alarm.
5. PPC Displays 882-887 identify flow upscale and compare alarm conditions.
6. The Rod Withdrawal Permissive indicating light will be off.

ACTIONS

1. If observations of FLOW (%) on APRM BARGRAPH displays at APRM ODAs, and FLOW COMPARE alarm indications on RBM ODA headers do not identify cause of the alarm, then perform the following:
   a. On the APRM ODAs, press ETC soft key to obtain TRIP STATUS soft key.
   b. Press TRIP STATUS soft key.
   c. Check FLOW UPSCALE ALARM status from the TRIP STATUS display.
   d. On RBM ODAs, press ETC soft key to obtain TRIP STATUS soft key.
   e. Press TRIP STATUS soft key.
   f. Check RECIRCULATION FLOW COMPARE status from TRIP STATUS display.
2. When the failed channel can be identified, then perform the following:
   a. Notify the Unit CRS.
   b. Bypass the affected APRM
   c. Confirm the FLOW REF OFF NORMAL annunciator clears.

From OI-18 for placing the APRM in Trip Condition:

INSTRUMENT NUMBER:	C51-APRM-Ch-1, 2, 3, 4
INSTRUMENT NAME:	APRMs
TS REFERENCE:	3.3.1.1; TRM Table 3.3.1.1-1.2a, b, c, d, f
TRIP CHANNEL:	Each APRM channel provides inputs to each of the four Voter channels. (APRMs are not trip channel specific)
TRIP SYSTEM:	Each APRM channel provides inputs to each of the four Voter channels
TRIP LOGIC:	Any two unbypassed APRMs in a tripped condition = Reactor Scram

Place APRM in tripped condition by: Placing APRM OPER/INOP mode selector switch in "INOP".

17. 216000 1

A reactor vessel instrument reference leg (with both level and pressure instruments) has CRD backfill in service.

A blockage of reference leg causes the instrument piping outside containment to equalize with CRD pressure.

Which one of the following completes the statements below?

The blockage will cause indicated level on the affected level instruments to ___(1)___.

An expected pressure alarm for these conditions would be ___(2)___.

A. (1) lower
   (2) A-04 (1-8) *Steam Line Lo Press A*

B. (1) lower
   (2) A-07 (3-2) *Reactor Press High*

C. (1) rise
   (2) A-04 (1-8) *Steam Line Lo Press A*

D. (1) rise
   (2) A-07 (3-2) *Reactor Press High*

Answer: B

K/A:
216000 NUCLEAR BOILER INSTRUMENTATION
A2 Ability to (a) predict the impacts of the following on the NUCLEAR BOILER INSTRUMENTATION ; 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)
02 Instrument line plugging

RO/SRO Rating: 2.9/3.0

Pedigree: Bank

Objective:
LOI-CLS-LP-001.2, Obj 5 - Explain the effect that the following will have on reactor vessel level and/or pressure indications: k) Instrument line plugging ***(LOCT)***

Reference: None

Cog Level: fundamental knowledge

Explanation:
Brunswick does not have a procedure for a plugged instrument line, so this question was only written to the predicting the effect on level indication and predicting the annunciator that would come in for the pressure instrument. The Chief examiner agreed with this testing philosophy.

Reference leg pressurization to CRD system (charging header) pressure causes pressure instruments to trend high, pressurizing reference leg causes level instrument DP to rise and indicated level to lower.

Distractor Analysis:

Choice A: Plausible because indicated level would be lowering and if the student correlates the same logic as the level then they would believe that pressure would also lower.

Choice B: Correct Answer, see explanation

Choice C: Plausible because reference leg pressure does go up but this creates a higher dp which causes level to indicate lower and if the student correlates the same logic as the level then they would believe that pressure would also lower.

Choice D: Plausible because reference leg pressure does go up but this creates a higher dp which causes level to indicate lower.

SRO Basis: N/A

Blocked Reference Leg

RTGB symptoms of complete blockage of reference leg with backfill in service:

1. D004A
   a. Half scram signal B-side
   b. Digital Feedwater System ignores level input from "A"
   c. Digital Feedwater System trouble alarm
   d. ADS Low Water Level alarm
   e. Reactor Pressure High alarm
   f. Reactor Pressure trends up, and then may drop back to normal, and reactor level remains the same on C32-LPR-R608
   g. B21-LI-R610 downscale (B21-LT-N036)
   h. B21-LI-R606A downscale (C32-LT-N004A)

18. 217000 1

Given the following plant conditions with RCIC in pressure control mode:

RCIC controller output	70%
E51-F022, Bypass to CST Vlv.	Throttled
RCIC Flow	300 gpm
RPV pressure	810 psig, slowly lowering
RCIC controller	Automatic set @ 300 gpm

Which one of the following identifies two independent actions that will stabilize RPV pressure?

The RO can throttle the E51-F022 in the ____(1)____ direction, or by ____(2)____ the RCIC Flow Controller auto setpoint.

A. (1) open
(2) lowering

B. (1) open
(2) raising

C. (1) close
(2) lowering

D. (1) close
(2) raising

Answer: A

K/A:
217000 REACTOR CORE ISOLATION COOLING SYSTEM (RCIC)
A4 Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8)
07 Reactor pressure

RO/SRO Rating: 3.9/3.8

Pedigree: Modified a bank question that was last used on the 2010-1 NRC exam.

Objective:
CLS-LP-016-A Obj. 17 - Describe how the following evolutions are performed during operation of the RCIC system: b.Adjusting RCIC flow in the reactor pressure control mode.

Reference: None

Cog Level: High

Explanation:
There are two ways to raise RPV pressure with the conditions given. One way is to open the 22 valve, thereby increasing the size of the hole and forcing the turbine to work less to deliver the same flowrate. The second is to lower the controller setpoint thereby causing the turbine to work less by forcing less flow through the same size hole.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because opening the F022 is correct and the student could have a misconception about the operation of the controller.

Choice C: Plausible because the student could have a misconception on the operation of the F022 valve and lowering the controller is correct.

Choice D: Plausible because if the operator ws trying to lower reactor pressure this would be correct.

SRO Basis: N/A

2010-1 Exam question:

. Given the following plant conditions with RCIC in pressure control mode:

RCIC controller output	70%
Bypass to CST Vlv, E51-F022	Throttled
RCIC Flow	300 gpm
RPV pressure	990 psig, slowly rising
RCIC controller	Automatic set @ 300 gpm

Which one of the following identifies two independent actions that will stabilize RPV pressure?

The RO can throttle the E51-F022 in the ____(1)____ direction, or by ____(2)____ the RCIC Flow Controller auto setpoint.

A. (1) open
(2) lowering

B. (1) open
(2) raising

C. (1) closed
(2) lowering

D. (1) closed
(2) raising

8. **THROTTLE OPEN** *BYPASS TO CST VLV, E51-F022,* **OR ADJUST** RCIC *FLOW CONTROL, E51-FIC-R600,* as necessary, to obtain the desired system parameters and reactor pressure. ☐

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19. 217000 2

Unit Two has inserted a manual scram.

Suppression Pool temperature is 90°F and rising due to HPCI/RCIC usage
Suppression Pool level is -25 inches
CST level is 21 feet

Which one of the following identifies:
(1) the lowest Suppression Pool Temperature that requires PCCP entry and
(2) the current suction source for the RCIC system?

A. (1) 96°F.
(2) CST.

B. (1) 96°F.
(2) Suppression Pool.

C. (1) 106°F.
(2) CST.

D. (1) 106°F.
(2) Suppression Pool.

Answer: A

K/A:
217000 REACTOR CORE ISOLATION COOLING SYSTEM (RCIC)
G2.04.04 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)

RO/SRO Rating: 4.5/4.7

Pedigree: New

Objective:
LOI-CLS-LP-016, Obj 5 - Given plant conditions, predict the RCIC System response to the following conditions:
g. High/low Suppression Pool water level.
h. Low CST level.

Reference: None

Cog Level: Hi

Explanation:
95°F is the entry condition for PCCP, 105°F is also an entry condition if RCIC is being run for surviellances. The CST is the normal suction for RCIC. It does transfer to the suppression pool on low level in the CST. The HPCI system has an auto transfer to the suppression pool on hi level in the suppression pool, but RCIC does not.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because >95 is correct and the HPCI system does transfer on high torus level.

Choice C: Plausible because >105 is correct if RCIC testing is in progress and the CST is correct.

Choice D: Plausible because >105 is correct if RCIC testing was in progress and the HPCI system does transfer on high torus level.

SRO Basis: N/A

The primary water supply to the RCIC System is the condensate storage tank (CST) through the normally open Condensate Storage Tank Suction Valve, E51-F010. In the event the CST level decreases to a predetermined level, the RCIC suction will automatically transfer to the suppression pool through normally closed Suppression Pool Suction Valves, E51-F029 and E51-F031.

Table 19-6 - HPCI Suppression Pool Suction Transfer Signals

Signal	Setpoint	Tech Spec
CST Level Low	23'5" elev. (3'5" tank level)	≥23'4" elev. (≥3'4" tank level)
Suppression Pool Level	-25"	≤-2'

PRIMARY CONTAINMENT CONTROL

PCCP-1

ENTRY CONDITIONS:

* SUPPRESSION POOL TEMP ABOVE 95°F <u>OR</u> ABOVE 105°F WHEN DUE TO TESTING
* DRYWELL AVERAGE AIR TEMP ABOVE 150°F
* DRYWELL PRESS ABOVE 1.7 PSIG
* SUPPRESSION POOL WATER LEVEL ABOVE - 27 INCHES (- 2 FEET & 3 INCHES)
* SUPPRESSION POOL WATER LEVEL BELOW - 31 INCHES (- 2 FEET & 7 INCHES)
* PRIMARY CTMT H_2 CONCENTRATION ABOVE 1.5%

PCCP-2

20. 218000 1

Which one of the following completes the statements below concerning operation of the SRVs?

___(1)___ causes annunciation of A-03 (1-10) *Safety / Relief Valve Open*.

Upon receipt of this alarm, at least one SRV ___(2)___ will be illuminated on the apron section of RTGB Panel P601.

A. (1) High temperature on recorder B2l-TR-6l4
   (2) red light ONLY

B. (1) High temperature on recorder B2l-TR-6l4
   (2) red and amber lights

C. (1) Activation of a SRV sonic detector
   (2) red light ONLY

D. (1) Activation of a SRV sonic detector
   (2) red and amber lights

Answer: D

K/A:
218000 AUTOMATIC DEPRESSURIZATION SYSTEM
A3 Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: (CFR: 41.7 / 45.7)
03 ADS valve acoustical monitor noise

RO/SRO Rating: 3.7/3.8

Pedigree: New

Objective:
LOI-CLS-LP-020, Obj 5 Describe the operation of the SRVs for both an overpressure condition and a manual/ADS actuation.

Reference: None

Cog Level: High

Explanation:
This alarm input is from the Sonic detectors and the alarm A-03 (1-1) Safety or Depress Vlv Leaking is from the temperature recorder. The red light indicates the valve is open and the amber light is a memory light.

Distractor Analysis:

Choice A: Plausible because high temperature causes a different alarm and the red light is illuminated.

Choice B: Plausible because high temperature causes a different alarm and both lights are illuminated

Choice C: Plausible because the sonic detector does cause the alarm and the red light is illuminated.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

DEVICE/SETPOINTS

SRV Sonic Detector Relay B21-74X-F	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-E	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-D	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-C	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-B	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-A	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-G	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-H	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-J	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-K	0.08 - 0.12 (volts)
SRV Sonic Detector Relay B21-74X-L	0.08 - 0.12 (volts)

DEVICE/SETPOINTS

Temperature Recorder B21-TR-614(SW1)	287-293°F (337-343°F for B21-F013C only)

21. 219000 1

Which one of the following states the normal electrical power supply to the following Unit One RHR Suppression Pool Cooling valves?

(1) 1-E11-F024A, RHR Torus Cooling Isolation Valve
(2) 1-E11-F028A, RHR Torus Spray Valve

A. (1) E5
(2) E5

B. (1) E5
(2) E7

C. (1) E7
(2) E5

D. (1) E7
(2) E7

Answer: B

K/A:
219000 RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE
K2 Knowledge of electrical power supplies to the following: (CFR: 41.7)
01 Valves

RO/SRO Rating: 2.5/2.9

Pedigree: New

Objective:
LOI-CLS-LP-017, Obj. 17b - List the normal and emergency power sources for the following: RHR MOVs

Reference: None

Cog Level: Memory

Explanation:
In order to establish suppression pool cooling the flowpath is through the F024 and F028 valves.
Power for the E11-F024A comes from E5 through 1XA MCC.
Power for the E11-F028A comes from E7 through 1XA-2 MCC.

Distractor Analysis:

Choice A: Plausible because both of these are Unit One Division I power to RHR Loop A valves.

Choice B: Correct Answer, see explanation

Choice C: Plausible because these are the reverse of the correct answer.

Choice D: Plausible because these are the Unit Two Division I power

SRO Basis: N/A

4160V EMERGENCY BUS E-1 ELECTRICAL LOAD LIST	0OI-50.1
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ATTACHMENT 7
Page 2 of 3

480V Motor Control Center 1XA Load Summary Sheet

Load: 480V Motor Control Center 1XA
Location: Reactor Building 20' N
Drawing Reference: F-30049
Upstream Power Source: **480V Substation E5**

CMPT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
DF7	E11-F024A (RHR Torus Cooling Isolation Valve) (TS 3.6.2.3, 3.6.1.3, 3.3.3.1)	Loss of load

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ATTACHMENT 11
Page 1 of 1

480V Motor Control Center 1-1XA-2 Load Summary Sheet

Load: 480V Motor Control Center 1-1XA-2
Location: Unit 1 Reactor Building 20' N
Drawing Reference: F-30049
Upstream Power Source: **480V Substation E7**

CMPT	LOAD DESCRIPTION	EFFECTS ON LOSS OF POWER
DF5	1-E11-F017A (RHR Outboard Injection Valve) (TS 3.3.3.1, 3.5.1, 3.5.2, 3.6.1.3)	Loss of load
DF3	1-E11-F015A (RHR Inboard Injection Valve) (TS 3.3.3.1, 3.5.1, 3.5.2, 3.6.1.3)	Loss of load
DG0	1-E11-F028A (RHR Suppression Pool Discharge Isolation Valve) (TS 3.3.3.1, 3.5.1, 3.5.2, 3.6.1.3, 3.6.2.3)	Loss of load

22. 223002 1

Unit Two is operating at power with DG3 under clearance for maintenance activities.
Bus E3 Master/Slave Breakers trip.

Which one of the following completes the statements below?

The ___(1)___ RWCU isolation valve auto closes.

Technical Specification LCO 3.6.1.3, Primary Containment Isolation Valves (PCIVs), states each PCIV, except ___(2)___, shall be OPERABLE.

A. (1) Inboard
(2) reactor building-to-suppression chamber vacuum breakers

B. (1) Inboard
(2) main steam isolation valves (MSIVs)

C. (1) Outboard
(2) reactor building-to-suppression chamber vacuum breakers

D. (1) Outboard
(2) main steam isolation valves (MSIVs)

Answer: C

K/A:
223002 PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF
G2.02.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations. (CFR: 41.10 / 43.2 / 45.13)

RO/SRO Rating: 3.1/4.2

Pedigree: Bank, last used on 2008 NRC Makeup Exam

Objective:
LOi-CLS-LP-014, Obj 9 - Given plant conditions, predict how the following will affect the RWCU System: m. Loss of AC power. (LOCT)

Reference: None

Cog Level: High

Explanation:
RWCU System outboard isolation logic contains contacts which opens when the NRHX outlet temperature exceeds its setpoint (135 °F, Only G31-F004 closes). The temperature sensing element is powered from 1AB-RX (2AB-RX) which is normally aligned to Division I AC. On a loss of this power, the Outboard, Division II, PCIS valve isolates. The RWCU Inboard and Outboard isolation valves are MOVs. The Inboard valve is powered from 480 Vac MCC XC, and the Outboard valve from 250 Vdc MCC XDB.

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

Distractor Analysis:

Choice A: Plausible because the inboard valves have lost power and if they were solenoid operated they would have closed.

Choice B: Plausible because the inboard valves have lost power and if they were solenoid operated they would have closed and the MSIV leakage not within limit is excepted in the action statements.

Choice C: Correct Answer, see explanation

Choice D: Plausible because MSIV leakage not within limit is excepted in the action statements.

SRO Basis: N/A

3.6 CONTAINMENT SYSTEMS

3.6.1.3 Primary Containment Isolation Valves (PCIVs)

LCO 3.6.1.3 Each PCIV, except reactor building-to-suppression chamber vacuum breakers, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,

When associated instrumentation is required to be OPERABLE per LCO 3.3.6.1, "Primary Containment Isolation Instrumentation."

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by PCIVs.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1.1, "Primary Containment," when PCIV leakage results in exceeding overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -----NOTE----- Only applicable to penetration flow paths with two PCIVs. ----- One or more penetration flow paths with one PCIV inoperable except for MSIV leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	8 hours

23. 230000 1

During a line break inside the drywell, plant conditions are:

RPV water level	200 inches
RPV pressure	800 psig
Drywell pressure	12 psig

Which one of the following completes the statements below?

In order to initiate Suppression Pool Sprays, operation of the "2/3 Core Height LPCI Initiation" keylock override switch is ___(1)___.

The Suppression Pool Spray valves ___(2)___ automatically close when drywell pressure lowers below 2.7 psig.

A. (1) required
   (2) will

B. (1) required
   (2) will NOT

C. (1) NOT required
   (2) will

D. (1) NOT required
   (2) will NOT

Answer: B

K/A:
230000 RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE
A3 Ability to monitor automatic operations of the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE including: (CFR: 41.7 / 45.7)
01 Valve operation

RO/SRO Rating: 2.5/2.9

Pedigree: New

Objective:
LOI-CLS-LP-300-K, Obj. 07 - Explain how the absence of a LPCI LOCA signal affects Drywell/Suppression Pool Spray operation.

Reference: None

Cog Level: Hi

Explanation:
Without a LPCI signal this keylock would be required to open the spray valves. If a LPCI signal is present and DW pressure lowers to less than 2.7 psig the valves will auto close, without the LPCI signal the close signal does not get energized. The 2/3 core height override (Lacka LOCA switch) and the closure logic is a common misconception in initial training.

Distractor Analysis:

Choice A: Plausible because the keylock is required and the valves will not auto close unless a LOCA is present.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the students typically have a misconception of the switch (Lack of LOCA) and the valves will close if a LOCA signal is present.

Choice D: Plausible because the students typically have a misconception of the switch (Lack of LOCA) and the valves will not auto close unless a LOCA is present.

SRO Basis: N/A

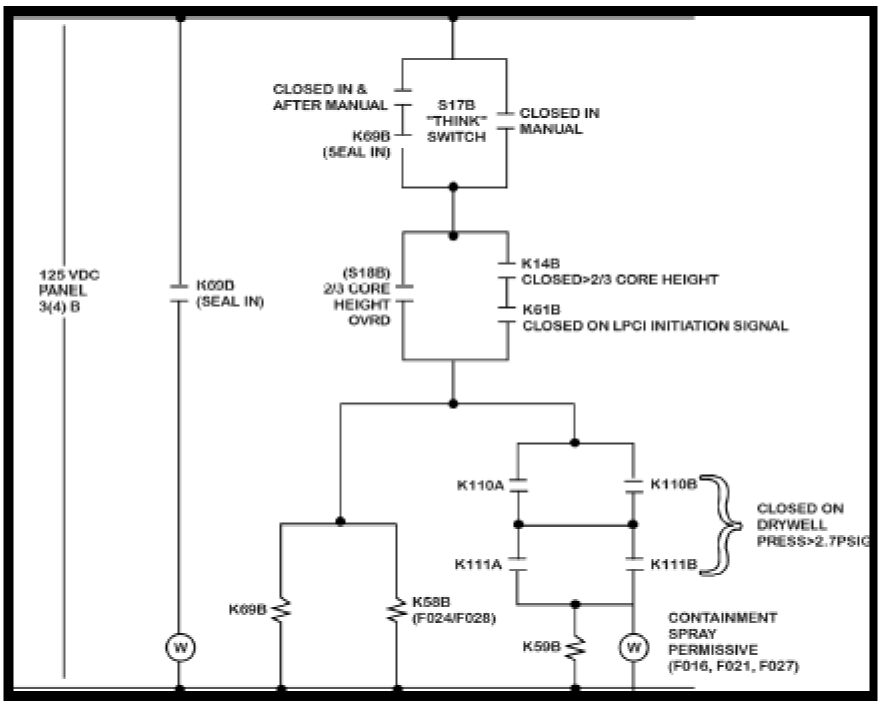
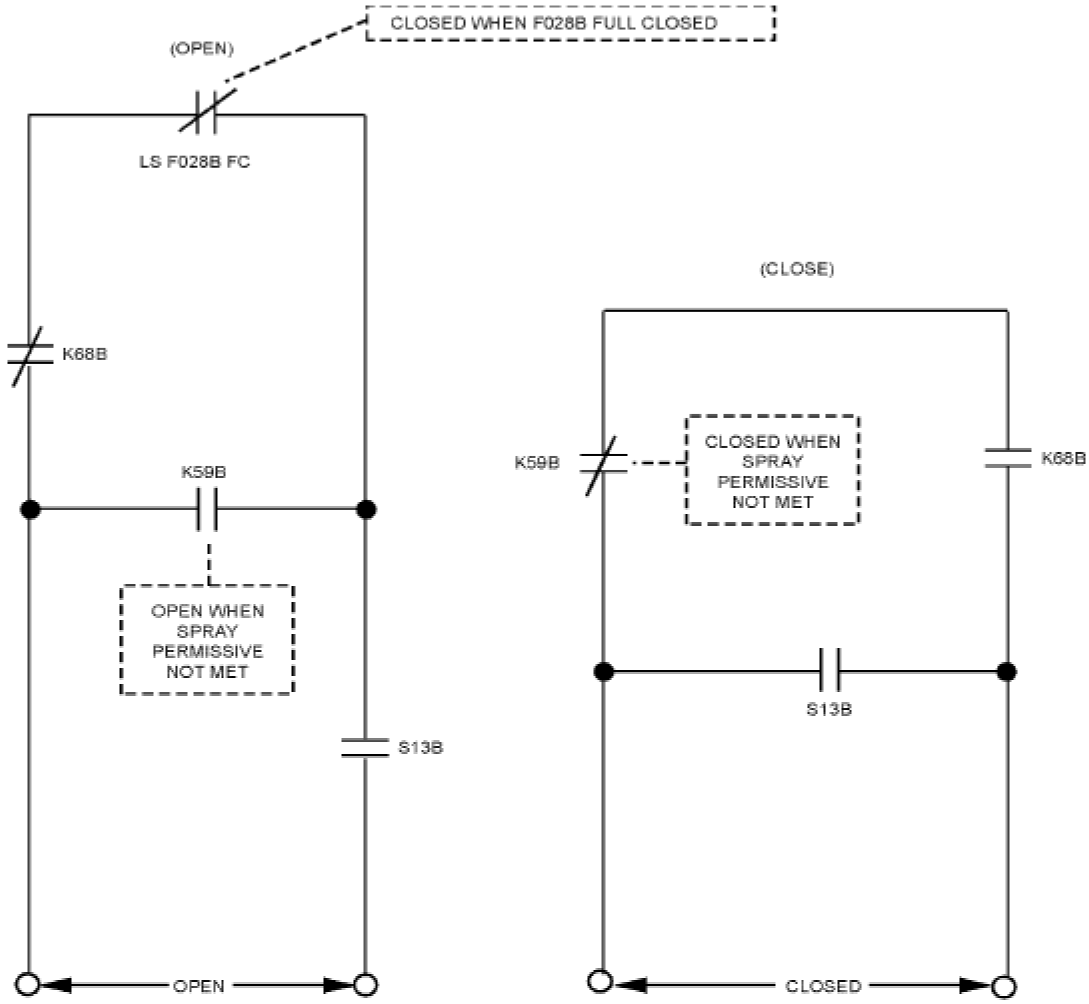


FIGURE 17-23
F027B Control Circuit



K59B - CONTAINMENT SPRAY PERMISSIVE

K68B - LPCI INITIATION

S13B - RTGB CONTROL SWITCH

24. 239002 1

Which one of the following identifies the loads that can be supplied by the Backup Nitrogen System?

A. Inboard MSIVs, SRV Accumulators, and Hardened Wetwell Vent Isolation valves.

B. Inboard MSIVs, Suppression Chamber to Drywell Vacuum Breakers, and Hardened Wetwell Vent Isolation valves.

C. SRV Accumulators, Reactor Building to Suppression Chamber Vacuum Breakers, and Hardened Wetwell Vent Isolation valves.

D. SRV Accumulators, Suppression Chamber to Drywell Vacuum Breakers, and Reactor Building to Suppression Chamber Vacuum Breakers

Answer: C

K/A:
239002 RELIEF/SAFETY VALVES
K1 Knowledge of the physical connections and/or cause effect relationships between RELIEF/SAFETY VALVES and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
05 Plant air systems

RO/SRO Rating: 3.1/3.3

Pedigree: Bank

Objective:
LOI-CLS-LP-046, Obj. 5 - List the pneumatic loads supplied by the Nitrogen Backup System

Reference: none

Cog Level: fundamental Knowledge

Explanation:
Following a Core Spray LOCA and subsequent containment isolation signal, the PNS (or RNA) supply to the Drywell will be isolated. Under these conditions, the Nitrogen Backup System supplies the SRV accumulators inside the Drywell and CAC-V16, CAC-V17, CAC-V7, and CAC-V216 valves outside the Drywell. No other loads inside the Drywell will be supplied by the Nitrogen Backup System. This ensures operability of the ADS Valves, the Reactor Building to Suppression Pool Vacuum Breakers, and the Wetwell Vents during all postulated accident conditions.

Distractor Analysis:

Choice A: Plausible because Inboard MSIVs are in the DW and have accumulators associated with them.

Choice B: Plausible because Inboard MSIVs are in the DW and have accumulators associated with them.

Choice C: Correct Answer, see explanation

Choice D: Plausible because the REactor Bldg vacuum breakers are supplied but the Torus to DW breakers are not.

SRO Basis: N/A

Following a Core Spray LOCA and subsequent containment isolation signal, the PNS (or RNA) supply to the Drywell will be isolated. Under these conditions, the Nitrogen Backup System supplies the SRV accumulators inside the Drywell and CAC-V16, CAC-V17, CAC-V7, and CAC-V216 valves outside the Drywell. No other loads inside the Drywell will be supplied by the Nitrogen Backup System. This ensures operability of the ADS Valves, the Reactor Building to Suppression Pool Vacuum Breakers, and the Wetwell Vents during all postulated accident conditions. The operator may override the PNS/RNA isolation and open both division Drywell PNS/RNA supply valves with the control switches to restore the pneumatic supply to all Drywell loads.

25. 239002 2

Which one of the following identifies the affect that a loss of E8 will have on the Unit Two Safety Relief Valve (SRV) system?

A. Inability to manually operate SRV's from the RTGB

B. Inability to manually operate SRV's from the RSDP

C. Loss of SRV position indication on the RTGB

D. Loss of SRV position indication on the RSDP

Answer: C

K/A:
239002 RELIEF/SAFETY VALVES
K6 Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES : (CFR: 41.7 / 45.7)
03 A.C. power

RO/SRO Rating: 2.7/2.9

Pedigree: Bank, last used on 2008 NRC exam

Objective:
CLS-LP-20, Obj. 15c. Given plant conditions, predict how ADS/SRVs will be affected by the following: Loss of AC power.

Reference: None

Cog Level: low

Explanation:
SRV position indication on the RTGB is powered thru the acoustic sensors which are powered from E6/E8.

Distractor Analysis:

Choice A: Incorrect. powered from 125 VDC

Choice B: Incorrect. powered from 125 VDC

Choice C: Correct Answer, see explanation

Choice D: Incorrect. powered from 125 VDC

SRO Basis: N/A

4.3.4 **AC Power** (Figure 20-8)

A loss of AC power to both E buses (loss of offsite power) with ADS valves open will close the ADS valves due to the trip of the low pressure ECCS pumps. As the diesel generators subsequently tie onto the E buses, the low pressure pumps will sequence on. If either E bus reenergizes and level is currently below LL3, the ADS valves will open. The ADS valves will open after the ECCS pumps sequence on only if level is below the LL3 initiation setpoint since, with the pumps not running, the K8A seal-in circuit is broken. The 83 second time delay circuit remains sealed-in, therefore resulting in immediate depressurization once adequate pump discharge pressure is sensed on the subchannel without the timer. In addition, a loss of E6/E8 results in a loss of power to the SRV acoustic sensors, requiring an alternate method to be used to determine ADS/SRV valve position (i.e., reactor pressure monitoring).

26. 245000 1

Unit Two is being shutdown for entry into the main generator for repairs.

Which one of the following completes the statement below concerning the flowpath for purging the Main Generator IAW 2OP-27.3, Generator Gas System Operating Procedure?

Carbon Dioxide exits through the ___(1)___ distribution tube in the main generator while
___(2)___ is admitted through the other distribution tube.

A. (1) upper
   (2) Hydrogen

B. (1) upper
   (2) Service Air

C. (1) bottom
   (2) Hydrogen

D. (1) bottom
   (2) Service Air

Answer: B

K/A:
245000 MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS
K1 Knowledge of the physical connections and/or cause effect relationships between MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
07 Plant air systems

RO/SRO Rating: 2.5/2.5

Pedigree: New

Objective:
LOI-CLS-LP-027.3, Obj 3 - Describe the flow path for the Main Generator Gas System

Reference: None

Cog Level: Fundamental Knowledge

Explanation:
Hydrogen is purged by CO_2 which is then purged by service air. the CO_2 enters through the bottom distribution tube and hydrogen is vented through the top distribution tube. Then spool pieces are removed and Service Air will use the bottom distribution tube while CO_2 is vented out of the hydrogen distribution tube.

Distractor Analysis:

Choice A: Plausible because the upper tube is correct and for startup CO2 does purge hydrogen.

Choice B: Correct Answer, see explanation

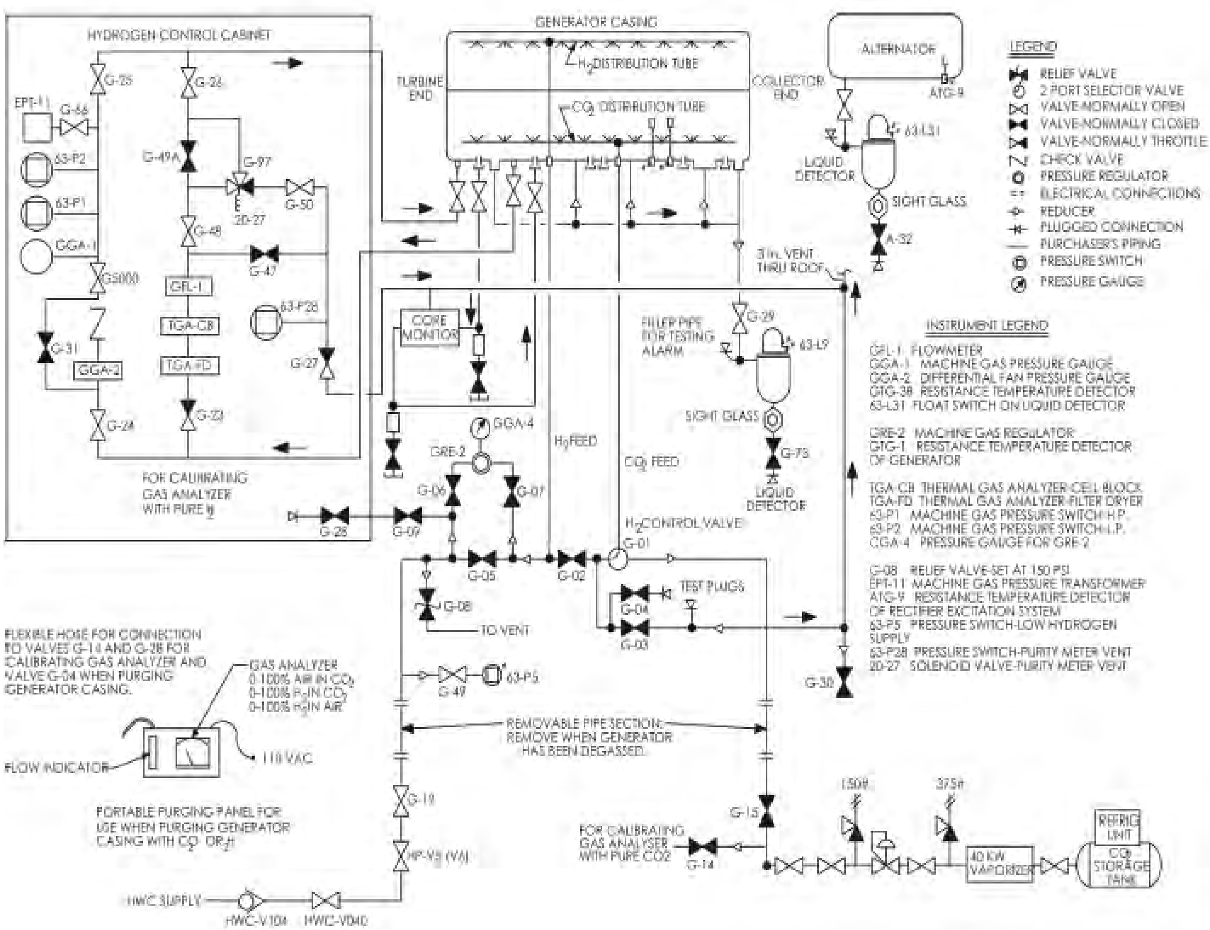
Choice C: Plausible because the bottom tube is the CO2 distribution tube and for startup CO2 does purge hydrogen.

Choice D: Plausible because the bottom tube is the CO2 distribution tube and Service Air is correct.

SRO Basis: N/A

4.1.3 Carbon Dioxide Purged by Air

Service Air is used to purge carbon dioxide from the generator casing if the casing must be entered for maintenance purposes. During this evolution Service Air is supplied to the generator casing by removing the CO2 spool piece and installing the Service Air adapter flange on the generator side of the supply line. Maintenance will also remove the hydrogen spool piece and install blank flanges on the generator and gas supply flanges. Service Air is admitted to the generator via the carbon dioxide distribution tube by placing the Flow Directional Valve, HC-G-01, in the carbon dioxide position and opening Hydrogen Purge Valves, HC-G-02 and HC-G-03, to vent the generator casing to the Turbine building roof. The portable gas analyzer is connected to Purging Sample Valve, HC-G-04, during this evolution to monitor percent carbon dioxide in air. Hydrogen Purge Valve, HC-G-03, is throttled during the purge to maintain generator gas pressure approximately 2 - 5 psig.



27. 256000 1

Unit Two is operating at 30% power when a Heater Drain (HD) Deaerator level controller failure results in HD Deaerator level rising to 62 inches.

Which one of the following completes the statements below?

MVD-LV-266 / 267, Deaerator Extraction Line MRVs, are ___(1)___.

EX-V11 / V12, 9th Stage Extraction Steam Non Return Valves, are ___(2)___.

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

Answer: B

K/A:

256000 REACTOR CONDENSATE SYSTEM

K4 Knowledge of REACTOR CONDENSATE SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7)

06 Control of extraction steam

RO/SRO Rating: 2.8/2.8

Pedigree: New

Objective:

LOI-CLS-LP-034, Obj 7 - Given plant conditions, describe the automatic feedwater heater level control actions for the following: c. High-High Feedwater Heater/Heater Drain Deaerator level

Reference: None

Cog Level: high

Explanation:

On hi-hi level (60 inches) in the deaerator the NRV close and the MRV open.

Distractor Analysis:

Choice A: Plausible because the MRVs do open and the normal position of the NRVs is open.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is the normal position of these valves.

Choice D: Plausible because the normal position of MRVs is closed and the NRVs do close

SRO Basis: N/A

Annunciation for High/Low level in the deaerator tank comes off level switches HD-LSH-95 (50”) and HD-LSL-94 (36”) on U1 or HD-PMC-REL04 (54"/36”) on U2. On a HI-HI level (60"), switch HD-LSHH-146 on U1 or HD-PMC-REL08 on U2 actuates and the following occurs (Figure 34-5 and 34-8):

- 9th stage Extraction Steam Non Return Valves (NRV) V-11 and V-12 close.
- 9th stage Moisture Removal Valves (MRV) MVD-266 and MVD-267 open.

This prevents water hammer by draining the remaining Extraction Steam to the Condenser prior to reopening of the NRV's.

28. 259001 1

Unit Two is performing plant heatup and pressurization with the reactor at 250 psig. Reactor Feed Pump (RFP) 2A indicates 185 RPM with the following status:

Suction valve is open
Recirc valve is open
Discharge valve is closed
UA-04 (1-2) *RFP A Turbine Tripped* is clear
HPU oil pressure is 275 psig
Reactor water level is 200 inches
A-07 (2-2) *Reactor Water Level High / Low* is in alarm

The operator depresses the RFPT A Start push button on XU-1 panel.

Which one of the following identifies how the 2A RFP will respond?

A. Rolls to 1000 RPM.

B. Rolls to 2450 RPM.

C. Remains at 185 RPM.

D. Trips on emergency shutdown logic.

Answer: A

K/A:
259001 REACTOR FEEDWATER SYSTEM
K5 Knowledge of the operational implications of the following concepts as they apply to REACTOR FEEDWATER SYSTEM : (CFR: 41.5 / 45.3)
03 Turbine operation

RO/SRO Rating: 2.8/2.8

Pedigree: New

Objective:
LOI-CLS-LP-32.3, Obj 5 - State the purpose/function of the following RFPT Governor controls: k. XU-2, RFPT RESET Pushbutton

Reference: None

Cog Level: High

Explanation:
The *RFPT A(B) START* pushbutton permissive light will be illuminated when all of the following conditions are met:

- Reactor feed pump has been reset at Panel XU-2
- Reactor feed pump suction pressure is normal
- RFPT A(B) HPU hydraulic pressure is greater than 175 psig
- *RFP A(B) SUCTION VLV, COD-V49(COD-V50),* is full open
- *RFP A(B) RECIRC VLV, FW-FV-V46(V47),* is full open

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because after rolling the turbine then the procedure has the operator roll the turbine to 2450 rpms.

Choice C: Plausible because if any of the conditions are changed this would be correct.

Choice D: Plausible because on high level (206 inches) would provide a trip and level is high just not high enough.

SRO Basis: N/A

Turbine startup is automatically performed by the Woodward controls. Depressing the START pushbutton automatically raises RFPT speed from 0 to 1000 RPM when all the following start permissives are satisfied:

1. Reactor feed pump suction valve open (COD-MO-V49/COD-MO-V50)

2. Reactor feed pump suction pressure is greater than 65 psig (COD-PS-3570/COD-PS-7584 & COD-PS-3571/COD-PS-7585)

3. Reactor feed pump minimum flow bypass valve open (FW-FV-46/FW-FV-47) and

4. Turbine reset (PS-16, Trip LO pressure > 95psig)

5. RFPT HPU hydraulic pressure greater than 175 psig

RFPT speed is then manually raised from 1000 rpm to 2550 RPM using a RAISE/LOWER switch on the XU-1 Panel. When RFPT speed reaches ≈ 2550 RPM (100 RPM above the lower limit of DFCS control) the Woodward feed pump controls are manually shifted to DFCS control using the MANUAL/DFCS switch on the XU-1 Panel. The RFP is manually controlled from the Control Room on panel H12-P603 by speed control stations C32-SIC-R601A and C32-SIC-R601B.

29. 259002 1

Given the following plant conditions on Unit One:

MODE 2 at 6% power
RPV pressure is 800 psig
SULCV is in Auto (40% valve demand)
Master Level Controller is in Manual set at 187 inches

A loss of UPS V7A results in blank displays on the Startup Level and Master Level Controllers.

Which one of the following identifies the response of the SULCV and the effect on reactor water level based on the above conditions?

The SULCV will:

A. close and the reactor will scram on low reactor water level.

B. open and the running Reactor Feed Pump will trip on high level.

C. remain at 40% valve demand position irrespective of reactor level changes.

D. change valve position as required to maintain reactor water level at 187 inches.

Answer: D

K/A:

259002 REACTOR WATER LEVEL CONTROL SYSTEM

K3 Knowledge of the effect that a loss or malfunction of the REACTOR WATER LEVEL CONTROL SYSTEM will have on following: (CFR: 41.7 / 45.4 to 45.8)

07 Reactor water level indication

RO/SRO Rating: 3.4/3.4

Pedigree: Bank

Objective:

LOI-CLS-LP-032-C, Obj. 7b - Given plant conditions, predict the effect a loss of or malfunction or misoperation of the DFCS will have on the following: b. RPV level and/or level indication (LOCT)

Reference: None

Cog Level: High

Explanation:

Feedwater control receives redundant power from DC distribution (panel 3B). Controllers are UPS only resulting in loss of operator interface with feedwater control system, but control system will function to maintain last demanded level.

Loss of UPS V7A will remove the capability to control the system from the specific control stations but will continue to function automatically from the logic control circuits. Loss of UPS V9A, 125VDC 3B, or 125VDC 3A will arm the associated High Level Trip circuit which would satisfy the two out of three logic needed to trip the Main and Reactor Feedpump Turbines on high level. Any power loss will initiate the "Feedwater Trouble" annunciator and the APP should be referenced.

Distractor Analysis:

Choice A: Plausible because on a loss of air the valve fails closed of on a loss of signal it may be thought that the level signal would be zero demand.

Choice B: Plausible because some valves do fail open.

Choice C: Plausible because some valves do fail as-is.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

Loss of UPS V7A/V8A will remove the capability to control the system from the specific control stations but will continue to function automatically from the logic control circuits. A loss of the UPS

4.3.1 Plant Air

The instrument air system supplies control air to the below listed valves, on a loss of instrument air the valves fail in the following position:

Valve Name	Failed Position
Feed Pump Recirc	Open
Condensate pump recirc	Open
Condensate SJAE Recirc flow	As-Is
Condensate booster pump recirc	Open
Heater drain pump level control	Open
CFD and CDD inlet and outlet	As - Is
SULCV	Closed
Condenser recirc (FW-177)	Closed

30. 261000 1

Unit One primary containment venting is being performed IAW 1OP-10, Standby Gas Treatment System Operating System with the following plant status:

1-VA-1F-BFV-RB, SBGT DW Suct Damper	Open
1-VA-1D-BFV-RB, Reactor Building SBGT Train 1A Inlet Valve	Closed
1-VA-1H-BFV-RB, Reactor Building SBGT Train 1B Inlet Valve	Closed

Which one of the following completes the statements below concerning the predicted SBGT response if drywell pressure reaches 1.9 psig?

1-VA-1F-BFV-RB ___(1)___.

Both 1-VA-1D-BFV-RB and 1-VA-1H-BFV-RB ___(2)___.

A. (1) auto closes
   (2) auto open

B. (1) auto closes
   (2) remain closed

C. (1) remains open
   (2) auto open

D. (1) remains open
   (2) remain closed

Answer: A

AK/A:

261000 STANDBY GAS TREATMENT SYSTEM

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

11 Primary containment pressure

RO/SRO Rating: 3.2/3.3

Pedigree: Bank

Objective:

LOI-CLS-LP-004.1, Obj 5 - List the signals and setpoints that will cause a Secondary Containment isolation

Reference: None

Cog Level: high

Explanation:

The filter train fans will automatically start on High Drywell Pressure.

The following actions occur: 1) SBGT Reactor Building suction dampers (1D-BFV-RB and 1H-BFV-RB) open, 2) SBGT DW Suct Damper (F-BFV-RB) closes.

The SBGT Train A/B Suction & Discahrge Valves on U1 do not auto open. These valves on U2 do auto open, so there could be a misconception on these valves (inlet vs suction dampers).

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because 1F does auto close and SBGT Train 1A/B Suction Valves (1C & 1E) on Unit One only do not auto open

Choice C: Incorrect since SBGT will auto realign from primary containment to the Reactor Building on system initiation

Choice D: Incorrect since SBGT will auto realign from primary containment to the Reactor Building on system initiation and SBGT Train 1A/B Suction Valves (1C & 1E) on Unit One only do not auto open

SRO Basis: N/A

2.1.6 Fan

A 100% capacity, heavy-duty, industrial type Fan and motor assembly is provided in each SBGT filter train. Each Fan will produce the required 2700 - 3300 scfm flow through its associated filter train.

Each Fan is driven by a direct-drive AC motor which is energized from a redundant and separate emergency power supply. The Unit 1 A and B Fans are powered from 480 VAC MCCs 1XE and 1XF respectively and Unit 2 A and B Fans from 2XE and 2XF.

The filter train fans may be operated manually from controls located at RTGB XU-51.

The filter train fans will automatically start if any of the following Secondary Containment isolation conditions exist: (Figure 10-2)

1. Low Reactor Water Level, LL #2
2. High Drywell Pressure
3. Reactor Building Ventilation Radiation (Figure 10-3)

3.2.6 Automatic

1. Upon receipt of an automatic initiation signal both trains of SBGT will start.

   Unit 1 ONLY

   The dampers associated with Unit 1 SBGT System will receive automatic open signals when an initiation signal is received EXCEPT for the train inlet and outlet dampers, (BFVs-1B,1C,1E,and 1G). Should these normally open dampers be manually closed locally via their CLOSE/OPEN pushbuttons, **they will NOT automatically reopen and the associated SBGT will not automatically start**.

31. 262001 1

The following sequence of events occur on Unit Two:

1156	Reactor scram due to high drywell pressure
1158	Off-site power is lost, DG4 locks out
1200	Reactor water level drops below LL2
1202	Bus E2 cross-tie breaker is placed to MAINT
1204	Reactor water level drops below LL3
1206	Bus E4 cross-tie breaker is placed to MAINT
1208	Reactor pressure lowers to 410 psig

Which one of the following identifies the earliest time that E4 is allowed to be energized from E2 IAW 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses?

A. 1206

B. 1208

C. 1214

D. 1216

Answer: D

K/A:

262001 A.C. ELECTRICAL DISTRIBUTION

A4 Ability to (a) predict the impacts of the following on the A.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)

02 Loss of coolant accident

RO/SRO Rating: 3.6/3.9

Pedigree: New

Objective:

LOI-CLS-LP-050-B, Obj. 13c - Given plant conditions, determine if the following breakers could be closed: E1 to E3 (or E2 to E4) cross-tie breakers

Reference: None

Cog Level: High

Explanation:

In the Maint position, there is a 10 minute trip on LOCA signal from either unit. The 10 minute timer cannot start until the breaker is placed to Maint since there is no DC control power with the switch in Norm. The bus E2 breaker 10 minute timer will not start when the switch is placed to maint since there is not yet a LOCA signal. the LOCA signal is received at 1204 which starts the 10 minute timer for the E2 breaker (which can be closed at 1214). The timer for the E4 breaker starts at 1206 so that breaker cannot be closed until 1216. (this predicts based on the LOCA when the ten minute timer starts (effect) and per the procedure when the breaker can be closed)

Distractor Analysis:

Choice A: Plausible because this would be correct if no LOCA signal was present.

Choice B: Plausible because this is 10 minutes from the LOOP, and when the LOCA signal is in from high DW pressure with low reactor pressure.

Choice C: Plausible because this is 10 minutes from the LOCA signal.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

LOSS OF ANY 4160V BUSES OR 480V E-BUSES	0AOP-36.1
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NOTE

- Placing SS-B, local cross-tie breaker (Control Selector Switch), to MAINT with a LOCA signal present initiates a 10 minute delay before breaker closure is permitted. ☐
- E bus cross-tie breaker keys (TEM30), will be needed to allow closing the breakers addressed in this section. ☐

32. 262002 1

The indications and status of the UPS System are:

	Primary Inverter	Standby Inverter
Load on UPS (DS10)	OFF	OFF
Load on Inverter (DS151)	OFF	ON
Load on Alternate (DS152)	ON	OFF
Alt Source Failure (DS11)	OFF	OFF
Manual Bypass Switch (S1)	NORM	BYP TEST

Which one of the following identifies the status of UPS System Loads?

A. de-energized.

B. powered from the primary inverter.

C. powered from the standby inverter.

D. powered from the alternate source.

Answer: D

K/A:

262002 UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.)

K6 Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : (CFR: 41.7 / 45.7)

03 Static inverter

RO/SRO Rating: 2.7/2.9

Pedigree: Bank

Objective:

LOI-CLS-LP-052, Obj. 5 - Given plant conditions, determine the lineup of the primary UPS, the Standby UPS, and their reserve sources. (LOCT)

Reference: None

Cog Level: High

Explanation:

The primary unit is in service with its output connected to the UPS distribution system. Its rectifier receives 480 VAC power from a Division I emergency distribution panel. A 250 VDC from DC Switchboard 1A (2A) is supplied in parallel with the rectifier output to power the inverter should the normal AC source be lost. The alternate AC source from the standby unit is available at the static transfer switch to pick up the loads if the inverter output is lost.

Distractor Analysis:

Choice A: Plausible because this would be true on a loss of AC under these conditions

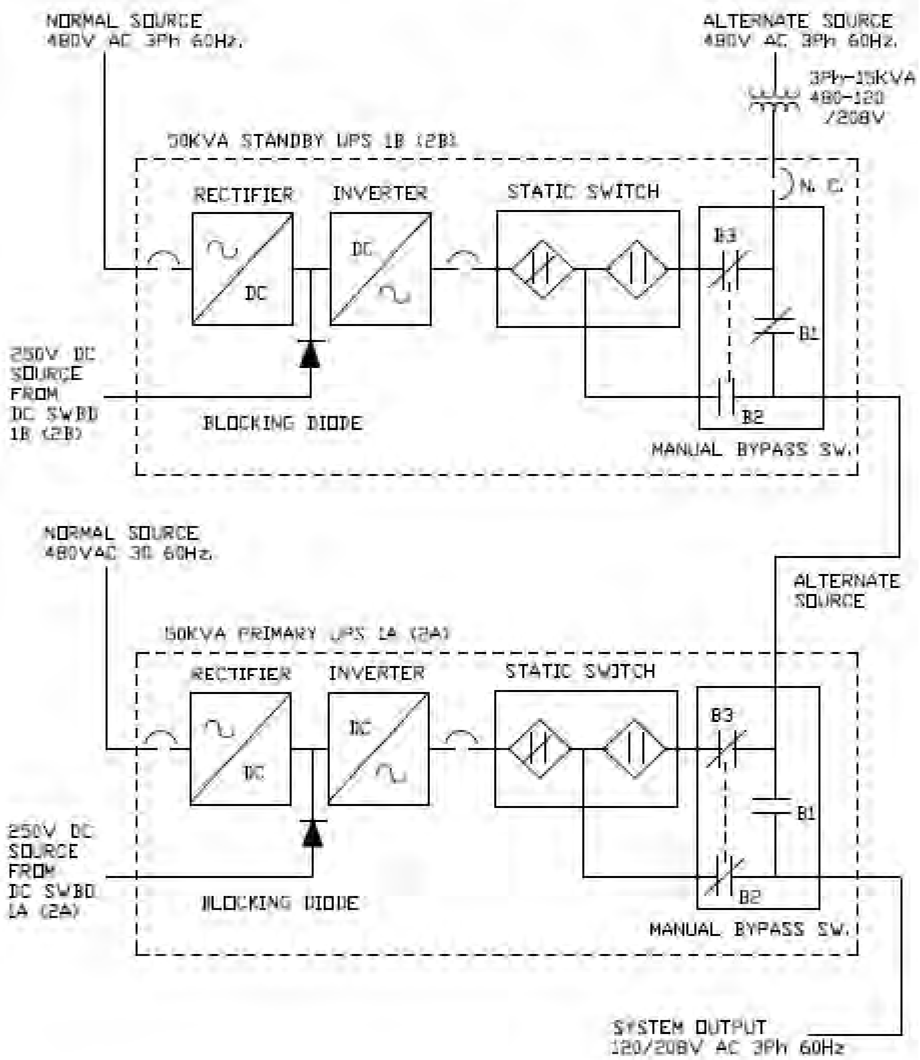
Choice B: Plausible because this is the normal power supply

Choice C: Plausible because this would be correct if the standby was in series with the primary as the original design was.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

FIGURE 52-7
Basic Vital UPS System



33. 263000 1

Which one of the following completes the statements below regarding 125/250 VDC Station Distribution?

In the equalize charge mode, the charger output voltage is at a ___(1)___voltage when compared to the float charge mode.

The 125 VDC batteries are sized to supply emergency power at a 150 amp rate for ___(2)___ hours.

A. (1) lower
(2) 8

B. (1) lower
(2) 10

C. (1) higher
(2) 8

D. (1) higher
(2) 10

Answer: C

K/A:

263000 D.C. ELECTRICAL DISTRIBUTION

A1 Ability to predict and/or monitor changes in parameters associated with operating the D.C. ELECTRICAL DISTRIBUTION controls including: (CFR: 41.5 / 45.5)

01 Battery charging/discharging rate

RO/SRO Rating: 2.5/2.8

Pedigree: New

Objective:

LOI-CLS-LP-051, Obj. 13 - Describe the location and operation of Battery Chargers 1B-1, 1B-2, 2B-1, and 2B-2 AC Power Transfer Switches.

Reference: None

Cog Level: Fundamental knowledge

Explanation:

The float mode voltage for the 125 VDC battery charger is ~135 volts while in equalize the charger output is ~140 volts. The design of the batteries is for 150 amps for 8 hours.

Distractor Analysis:

Choice A: Plausible because the student may have a knowledge deficiency on which (float vs equalize) value is for equalize and the 8 hours is correct.

Choice B: Plausible because the Caswell Beach batteries are rated for 10 hours and the student may have a knowledge deficiency on which (float vs equalize) value is for equalize.

Choice C: Correct answer, see explanation.

Choice D: Plausible because higher is correct and the Caswell Beach batteries are rated for 10 hours.

SRO Basis: N/A

2.0 COMPONENT DESCRIPTION/DESIGN DATA

2.1 Battery Capacity Ratings

All of the battery systems (with the exception of the Caswell Beach Microwave) have a design Ampere-Hour capacity rating which defines the batteries expected lifetime, in hours, based upon a given continuous loading, in amperes. It should be noted that this is merely a reference number and that battery lifetime is shortened if it is discharged at a higher rate or lengthened if discharged at a lower rate. The individual battery capacities are:

BATTERY SYSTEM	AMP-HOUR RATING
125/250 VDC Station (each division)	1200 AMP-HOURS at a 150 amp rate for 8 hours
24/48 VDC Station (each division)	600 AMP-HOURS at a 75 amp rate for 8 hours
125 VDC Caswell Beach	200 AMP-HOURS at a 20 amp rate for 10 hours

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There is no direct indication of the status of the battery charger; i.e., whether it is in the float charge or equalizer charge mode. If in the float charge mode the volt meter should read approximately 135 VDC. If in the equalizer charge mode the meter should read approximately 140 VDC.

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34. 264000 1

During an ATWS on Unit Two, RPV level is being controlled at Top of Active Fuel.

The RHR pumps have been overridden OFF.

A fault then occurs on Bus 2C which results in loss of Bus E4.

Which one of the following identifies the RHR pump response as DG4 re-energizes Bus E4?

A. RHR pumps 2B and 2D both remain overridden off.

B. RHR pumps 2B and 2D both restart 10 seconds later.

C. RHR pump 2D restarts 10 seconds later, RHR pump 2B remains off.

D. RHR pump 2B restarts 10 seconds later, RHR pump 2D remains off.

Answer: D

K/A:

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)

05 Load shedding and sequencing

RO/SRO Rating: 3.2/3.5

Pedigree: Bank

Objective:

LOI-CLS-LP-17, Obj 7 - Given plant conditions, determine if the RHR System should automatically initiate in the LPCI mode.

Reference: none

Cog Level: High

Explanation:

The RHR System will automatically start in the LPCI mode of operation in response to either of two initiation signals: reactor vessel low level (LL3) or drywell high pressure with reactor vessel low pressure.
All RHR Pumps automatically start 10 seconds from receipt of the initiation signal if the Emergency busses are energized (off-site power available). If off-site power is not available, the pumps automatically start 10 seconds from the time the Emergency Diesel Generators re-energize its' bus.

Distractor Analysis:

Choice A: Plausible because they were overridden prior to DG4 re-energizing E4.

Choice B: Plausible because if E2 is also re-energized this would be correct.

Choice C: Plausible because if it is determined that E1-2A, E2-2B, E3-2C, & E4-2D are the power supplies then this would be correct.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

The power supply for the pumps, along with the associated diesel generator and power supply division, is listed below (also see Figure 17-2B):

RHR Pump	1A/2A	1B/2B	1C/2C	1D/2D
Power Source	E3	E4	E1	E2
Diesel	#3	#4	#1	#2
Division	I	II	I	II

Pump start sequencing is provided by the load sequencing logic that is automatically provided during auto initiation conditions. No manual operation is required for emergency pump starts.

After an initiation signal is received the following actions will occur:

- all four RHR pumps will start 10 seconds after power is available to the E-buses.

35. 271000 1

Unit Two is operating at rated power with Offgas Train A in full load.
A tube rupture occurs inside the Offgas Aftercondenser.

Which one of the following parameters will lower/diminish in response to this event?

A. Aftercondenser Outlet Temperature

B. Offgas Filter differential pressure

C. Main condenser vacuum

D. Aftercondenser Level

Answer: C

K/A:
271000 OFFGAS SYSTEM
K3 Knowledge of the effect that a loss or malfunction of the OFFGAS SYSTEM will have on following: (CFR: 41.5 / 45.3)
01 Condenser vacuum

RO/SRO Rating: 3.5/3.5

Pedigree: New

Objective:
LOI-CLS-LP-030, Obj 6 - Given the necessary plant conditions, describe the effect that each of the following will would have on the Condenser Air Removal/Augmented Offgas System: (LOCT) j. (LOCT) Loss or failure of the Condensate System

Reference: None

Cog Level: hi

Explanation:
The reason is simply due to the “back-pressure” placed on the Offgas stream flow felt all the way back through the Offgas system to the Condenser as a result of the flooded condenser.

Distractor Analysis:

Choice A: Plausible because when the condenser floods, cooling water recirculation (i.e., heat transfer of BTUs away from the condenser) essentially stops; the “pool” of water in the shell-side becomes ineffective, causing a rise in the Offgas outlet temperature (i.e., less cooling of the Offgas stream through the condenser shell)

Choice B: Plausible because the d/ps will increase not lower.

Choice C: Correct Answer, see explanation

Choice D: Plausible because level will rise, not lower.

SRO Basis: N/A

Condensate System

The condensate system provides the heat sink for the steam packing exhauster, aftercondenser, and SJAE intercondenser. Condensate first flows through the Steam Packing Exhauster and then, via parallel paths, flows through the Inner and After Condensers. A loss of condensate flow through the heat exchangers will result in reduced SJAE efficiency, decrease in condenser vacuum, increased moisture carryover to the recombiner causing a decrease in recombiner efficiency, increased aftercondenser outlet temperature, increased filter d/p's and increased moisture carryover from the steam packing exhauster to the Main Stack.

36. 290001 1

Which one of the following identifies the action that is required to be taken in response to annunciator UA-05 (1-9) *Fan Clg Unit CS Pump Rm A Inl Press Lo*?

IAW the APP, the reactor operator will open:

A. SW-V105, Nuc SW Supply Vlv.

B. SW-V101, Conv SW Supply Vlv.

C. SW-V143, Well Water Supply Vlv.

D. SW-V117, Nuc SW to Vital header Vlv.

Answer: D

K/A:
290001 SECONDARY CONTAINMENT
G2.04.31 Knowledge of annunciator alarms, indications, or response procedures. (CFR: 41.10 / 45.3)

RO/SRO Rating: 4.2/4.1

Pedigree: New

Objective:
LOI-CLS-LP-043, Obj 9 - Describe the operation of the Core Spray and RHR Room Coolers including any limitations on operation.

Reference: None

Cog Level: hi

Explanation:
These alarms are for low pressure in the vital header which supplies cooling to the ECCS Room Coolers. The APP requires opening the SW-V111 or SW-V117 valve. The other valves are SW valves associated with the RHR SW system.

Distractor Analysis:

Choice A: Plausible because this is a SW valve associated with the RHR SW system and the valves are located in close proximity to the V117 and are operated for initiating RHR SW.

Choice B: Plausible because this is a SW valve associated with the RHR SW system and the valves are located in close proximity to the V117 and are operated for initiating RHR SW.

Choice C: Plausible because this is a SW valve associated with the RHR SW system and the valves are located in close proximity to the V117 and are operated for initiating RHR SW.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

FAN CLG UNIT CS PUMP RM B INL PRESS LO

(Fan Cooling Unit Core Spray Pump Room B Inlet Pressure Low)

AUTO ACTIONS

NONE

CAUSE

1. Auto start of Core Spray Pump B Room Cooler or RHR Pump Room Cooler B without vital service water header in operation.
2. If the vital service water header is being supplied from the service water nuclear header, low service water nuclear header pressure.
3. If the vital service water header is being supplied from the service water conventional header, low service water conventional header pressure.
4 Improper valve alignment.
5. Circuit malfunction.

OBSERVATIONS

1. Service water nuclear header pressure, as indicated on SW-PI-143-1 on RTGB Panel XU2, decreasing.
2. Service water conventional header pressure, as indicated on SW-PI-131-1 on RTGB Panel XU2, decreasing.

ACTIONS

1. If vital service water header is not in operation with a room cooler operating, then place vital service water header in operation (SW-V111 or SW-V117).

37. 295001 1

A reactor recirc pump has tripped on Unit Two.

Which one of the following completes the statement below for determining stability region compliance?

The primary indication of total core flow is determined using:

A. Core Support Plate Delta-P.

B. PPC Point U2NSSWDP (WDP).

C. Total Core Flow recorder (R613).

D. PPC Point U2CPWTCF (WTCF).

Answer: D

K/A:

295001 PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION

AK1 Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION:(CFR: 41.8 to 41.10)

02 Power/flow distribution

RO/SRO Rating: 3.3/3.5

Pedigree: New

Objective:

LOI-CLS-LP-302-C, Obj 7 - Given plant conditions and AOP-03.0, determine the required supplementary actions. *(LOCT)*

Reference: None

Pedigree: Bank

Cog Level: Fundamental Knowledge

Explanation:

Process Computer Point U2CPWTCF when validated, is the primary indication of total core flow, and should be used for stability region compliance. The recorder will read lower than the PPC points. If WTCF is not validated then WDP can be used. If WTCF and WDP are not valid then

Distractor Analysis:

Choice A: Plausible because this indication is used if WTCF and WDP are not available. Attachment in the AOP to correlate this to core Flow.

Choice B: Plausible because this indication is used if WTCF is not available.

Choice C: Plausible because this indication is available on the P603 panel.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

NOTE

- PPC Point U2CPWTCF, when validated, is the primary indication of total core flow, and should be used for stability region compliance. If U2CPWTCF is invalid, U2NSSWDP or Attachment 1, Estimated Total Core Flow vs. Core Support Plate Delta-P, may be used as alternate indications for total core flow. .... ☐
- As the instability region is approached, PPC Point B018, Total Core Flow, and B21-R613 (Core Δ Pressure/Core Flow) recorder, located on Panel P603, will read lower than PPC Point U2CPWTCF or U2NSSWDP. .... ☐
- Operation outside the analyzed regions of the power to flow map should be minimized. .... ☐

38. 295003 1

Both Units were operating at rated power when ALL switchyard PCB position indications turn green.

Diesel Generator status:

DG1	Running loaded
DG2	Under clearance
DG3	Running loaded
DG4	Tripped on low lube oil pressure

Which one of the following identifies the AOP(s) that Unit One and Unit Two are required to perform?

Unit One is required to perform ___(1)___.
Unit Two is required to perform ___(2)___.

A. (1) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses
   (2) 0AOP-36.2, Station Blackout

B. (1) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses
   (2) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses

C. (1) 0AOP-36.2, Station Blackout
   (2) 0AOP-36.2, Station Blackout

D. (1) 0AOP-36.2, Station Blackout
   (2) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses

Answer: B

K/A:
295003 PARTIAL OR COMPLETE LOSS OF A.C. POWER
AA2 Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : (CFR: 41.10 / 43.5 / 45.13)
05 Whether a partial or complete loss of A.C. power has occurred

RO/SRO Rating: 3.9/4.2

Pedigree: Bank, last used on 2010-1 NRC exam

Objective:
LOI-CLS-LP-303-A, Obj 1 - Given plant conditions and control room indications, determine if AOP 36.2, Station Blackout Procedure, should be entered.

Reference: none

Cog Level: High

Explanation:
This meets the KA because the student will have to determine that all green lights is a LOOP on BOTH Units then determine that neither unit is in Station Blackout since each unit has a running and loaded D/G available.

Switchyard PCB green position indication shows all PCB are OPEN, which indicates Loss of ALL offsite power.

Distractor Analysis:

Choice A: Plausible because examinee could misdiagnose unit specific D/G availability

Choice B: Correct Answer, see explanation

Choice C: Plausible because examinee could misdiagnose unit specific D/G availability

Choice D: Plausible because examinee could confuse a "loss of off-site power on both units" as an entry condition for AOP-36.2 on both units

SRO Basis: N/A

From AOP-36.1:

1.0 SYMPTOMS

1.1 Loss of Off-site Power

1.1.1 SAT de-energized

1.1.2 Buses B, C, and D undervoltage

1.1.3 Indication of all four diesel generators running

1.2 Loss of E Bus

1.2.1 One 4160V or 480V E Bus undervoltage

1.2.2 Loss of one RPS bus (half-Scram signal)

1.2.3 Partial loss of instrumentation powered from Emergency 120 VAC

1.2.4 Indication of one diesel generator running

1.3 Loss of One BOP Bus

1.3.1 One 4160V BOP bus undervoltage

1.3.2 Reactor recirculation pump trip

1.3.3 Indication of one or two diesel generators running

From AOP-36.2:

1.0 SYMPTOMS

1.1 SAT deenergized

1.2 Buses B, C, and D undervoltage

1.3 Bus E1 and E2 (E3 and E4) undervoltage

1.4 No diesel generators running and loaded on one or both units

2.0 AUTOMATIC ACTIONS

2.1 Reactor scram

2.2 Groups 1, 2, 6, and 10 isolate

2.3 Groups 3 and 8 isolate with the DC powered outboard isolation valves only.

2.4 Reactor Building HVAC trips, but does **NOT** isolate until power is available to the damper solenoid valves.

2.5 The following DC oil pumps start on low header pressure:

- RFPTs
- Reactor Recirc M-G Sets
- Main Turbine
- Hydrogen Seal Oil

39. 295004 1

Which one of the following identifies how the manually initiated, automatically executed, fast bus transfer capability is affected following a loss of 125V DC Panel 9A?

The fast bus transfer will ____(1)____ if attempted for 4 KV Bus 1B.
The fast bus transfer will ____(2)____ if attempted for 4 KV Bus 1C.

A. (1) occur
   (2) NOT occur

B. (1) occur
   (2) occur

C. (1) NOT occur
   (2) NOT occur

D. (1) NOT occur
   (2) occur

Answer: A

K/A:
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)
03 A.C. electrical distribution

RO/SRO Rating: 3.4/3.6

Pedigree: Bank, last used on the 2010-1 NRC exam

Objective:
CLS-LP-50.1 Obj 7 - Given plant conditions, predict the effect a loss of DC control power will have on the 4160 VAC System.

Reference: None

Cog Level: Hi

Explanation:

Distractor Analysis:
BOP Bus 1B has AUTO control power transfer capability where 1C and 1D do not.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the auto transfer of control power will occur on 1B, but will not on 1C and D. Recent plant mods have removed some of the auto transfer capabilities on some of the DC control power arrangements (E-busses require a manual transfer of control power).

Choice C: Plausible because the auto bus transfer will not occur on 1C and D, but it will on 1B. Recent plant mods have removed some of the auto transfer capabilities on some of the DC control power arrangements.

Choice D: Plausible because the examinee may have the logics reversed.

SRO Basis: N/A

3	Switchgear Bus 1B Control Power	1. Automatic Bus Transfer to alternate power, Panel 10A, ckt. 11. **Note:** This control power feeds the following loads: a. 1A and 1B Recirc VFD Supply breakers. • Breaker closing and trip circuit. • Breaker local and remote indication. • *ATWS Trip Logic. *This load has alternate power from Panels 3A and 3B. If power swaps to the alternate source, annunciator A-06 6-4 and/or A-06 6-5 would be received. The local lockout switch must be reset to clear the alarms. b. Incoming Line Breakers from the UAT and SAT, control and indication. c. BOP 4KV Bus B undervoltage and over-current protection. d. Unit 1 1B to 4KV/480V PDC XFMR Brkr, A1E • Breaker closing and trip circuit. • Breaker local and remote indication.

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PANEL 9A Reference Drawing: LL-30024-12	LOCATION: Unit 1 Turbine Building, 20', Switchgear area	NORMAL SUPPLY: Switchboard 1B	ALTERNATE SUPPLY: Switchboard 1A (Mechanical Interlock)

CKT #	LOAD	EFFECT
4	Bus Common A Control Power	1. Loss of control power to 4KV loads on Common A. 2. Loss of 4KV breaker operation, manual or automatic. 3. 4KV breakers fail as is. 4. Loss of breaker indication locally and on the RTGB.
5	Generator Field Rectifiers High Temperature Alarm Circuit	1. Receive annunciator 1-UA-02 5-9. 2. Local high temp lights will not illuminate.
6	Switchgear Bus 1C Control Power	1. Loss of control power to 4KV loads on Bus 1C. 2. Loss of 4KV breaker operation, manual or automatic. 3. Loss of 4KV protective functions for Bus 1C and 4KV loads. 4. Bus 1C or 1D will not automatically 'dead bus' auto transfer from the UAT to SAT. (Loss of high speed transfer sync. circuit) 5. Bus 1C or 1D cannot be manually initiated, auto executed, fast bus transferred. (Loss of high speed transfer sync. circuit) 6. Loss of breaker indication locally and on the RTGB. 7. Div II Loss of BOP Bus DG auto start signal INOP. 8. Div II SAT secondary winding undervoltage LOOP signal INOP.

40. 295005 1

Which one of the following completes the statements below concerning the bases for the Reactor Protection System initiating a scram on Turbine Trip / Turbine Stop Valve closure?

Anticipates the pressure, neutron flux, and heat flux rise due to the ___(1)___ in voids.

The scram reduces the energy that must be absorbed to ensure the ___(2)___ safety limit is not exceeded.

A. (1) increase
(2) MCPR

B. (1) increase
(2) Reactor Coolant System pressure

C. (1) decrease
(2) MCPR

D. (1) decrease
(2) Reactor Coolant System pressure

Answer: C

K/A:
295005 MAIN TURBINE GENERATOR TRIP
AK1 Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP : (CFR: 41.8 to 41.10)
01 Pressure effects on reactor power

RO/SRO Rating: 4.0/4.1

Pedigree: Bank

Objective:
LOI-CLS-LP-026, Obj 5 - Describe the operation of the following Main Turbine related components:
a. Turbine Stop Valves.

Reference: None

Cog Level: Fundamental knowledge

Explanation:
The purpose of this scram is to anticipate the pressure, neutron flux, and heat flux increases that would result from closure of the Turbine Stop Valves. The pressure increase from turbine stop valve closure would increase reactivity due to steam void collapse and acts to protect the MCPR safety limit.

Distractor Analysis:

Choice A: Plausible because they may think backwards as a reduction in power will cause an increase in voids and MCPR is correct.

Choice B: Plausible because they may think backwards as a reduction in power will cause an increase in voids and there is an increase in pressure.

Choice C: Correct Answer, see explanation

Choice D: Plausible because decrease is correct and there is an increase in pressure.

SRO Basis: N/A

3.1.2 Turbine Trip/Turbine Stop Valves, Setpoint < 90% Full Open

The purpose of this scram is to anticipate the pressure, neutron flux, and heat flux increases that would result from closure of the Turbine Stop Valves. The pressure increase from valve closure would increase reactivity due to steam void collapse, resulting in higher power and heat flux.

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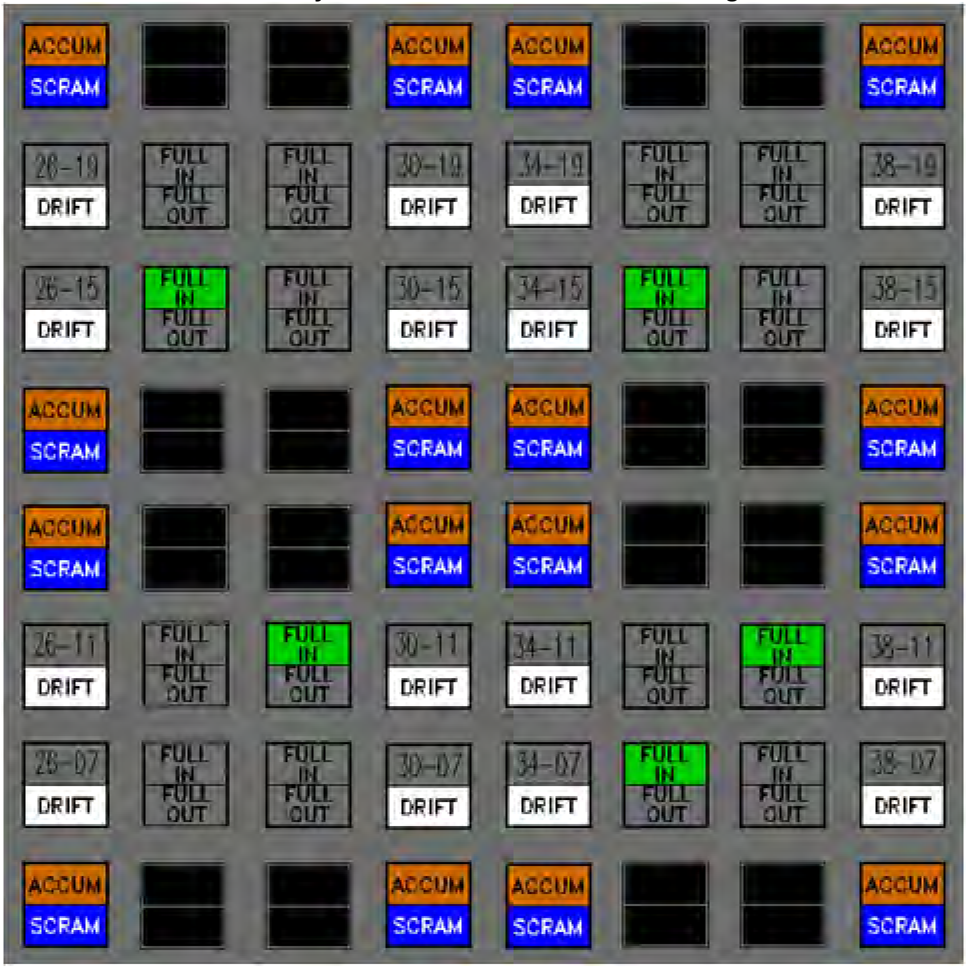
From TS Bases B3.3.1.1

8. Turbine Stop Valve—Closure

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve—Closure Function is the primary scram signal for the turbine trip event analyzed in Reference 2. For this event, the reactor scram reduces the amount of energy required to be absorbed and ensures that the MCPR SL is not exceeded.

41. 295006 1

Unit Two was manually scrammed with the following indications.



Which one of the following completes the statements below?

___(1)___ ATWS has occurred.

The reactor ___(2)___ remain shutdown under all conditions without boron.

A. (1) A hydraulic
   (2) will

B. (1) A hydraulic
   (2) will NOT

C. (1) An electrical
   (2) will

D. (1) An electrical
   (2) will NOT

Answer: B

K/A:
295006 SCRAM
AK1 Knowledge of the operational implications of the following concepts as they apply to SCRAM: (CFR: 41.8 to 41.10)
02 Shutdown margin

RO/SRO Rating: 3.4/3.7

Pedigree: New

Objective:
LOI-CLS-LP-003, Obj. 14 - Given plant conditions and control room indications, determine whether a reactor scram has actuated properly.
Reference: None

Cog Level: Hi

Explanation:
A hydraulic ATWS is indicated by the blue scram lights being energized but a failure of the rods to insert. This makes the electrical ATWS choices wrong.
Since greater than 10 rods are not fully inserted (at position 02 or greater) the reactor will not remain shutdown under all conditions without boron

Distractor Analysis:

Choice A: Plausible because this is a hydraulic ATWS and if one more rod was inserted then this would be correct.

Choice B: Correct Answer, see explanation

Choice C: Plausible because if the blue scram lights were not lit this would be correct and if one more rod was inserted then this would be correct.

Choice D: Plausible because if the blue scram lights were not lit this would be correct

SRO Basis: N/A

SHUTDOWN WITHOUT BORON
ONLY ONE CONTROL ROD NOT FULLY INSERTED
NO MORE THAN 10 CONTROL RODS WITHDRAWN TO POSITION 02 AND NO CONTROL ROD WITHDRAWN BEYOND POSITION 02
AS DETERMINED BY REACTOR ENGINEERING

42. 295008 1

Following a reactor scram on Unit One, plant conditions are:

Reactor water level	220 inches, rising
Reactor pressure	350 psig, steady
Drywell ref leg temp	205°F, steady

Which one of the following identifies the indicated level (on level indicator N027A/B) that corresponds to the bottom of the Main Steam Lines?

(Reference provided)

A. 240 inches

B. 245 inches

C. 250 inches

D. 255 inches

Answer: D

K/A:

295008 HIGH REACTOR WATER LEVEL

AA2 Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13)

01 Reactor water level

RO/SRO Rating: 3.9/3.9

Pedigree: Bank question

Objective:

LOI-CLS-LP-300C, Obj 10 - Given plant conditions and the Reactor Scram Procedure, determine the required operator actions. (LOCT)

Reference: 0EOP-01-UG, Att. 6, Figure 21

Cog Level: high

Explanation:

Using the upper line (Ref Leg Temp $\geq$200°F) of MSL elevation correction graph at 350 psig, corrected MSL elevation is at 255" requiring closure of MSIVs.

Distractor Analysis:

Choice A: Plausible because this is symetrical with the other plausibe answers.

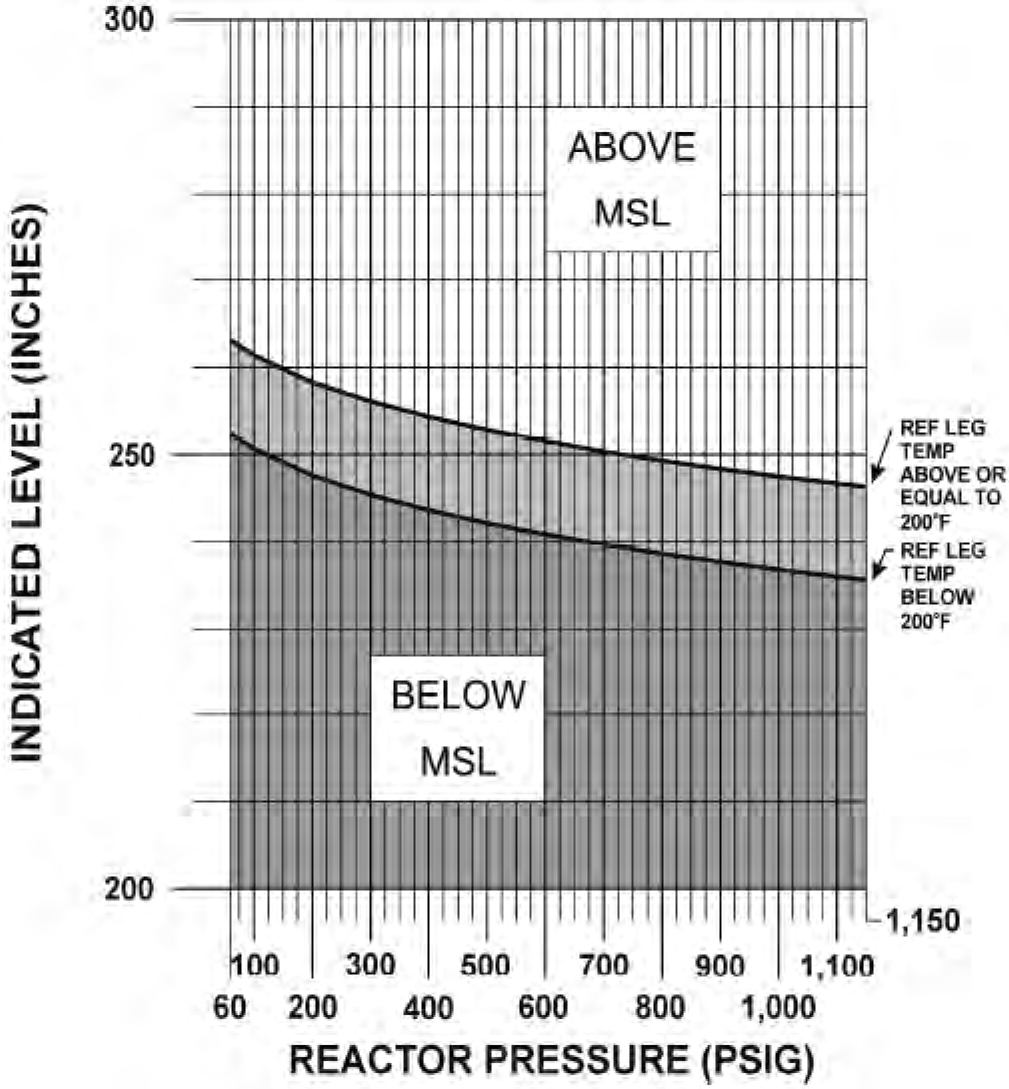
Choice B: Plausible because this would be correct if ref leg was <200°F.

Choice C: Plausible because this is the normal level of the MSL.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

FIGURE 21
Reactor Water Level at MSL
(Main Steam Line Flood Level)



WHEN REACTOR PRESSURE IS LESS THAN
60 PSIG, USE INDICATED LEVEL.
MSL IS +250 INCHES.

43. 295014 1

Unit Two is operating at 60% when Reactor Recirculation Pump 2A speed begins to slowly rise.

Which one of the following identifies an immediate action required IAW 2AOP-03.0, Positive Reactivity Addition?

A. Depress 2A VFD Stop pushbutton.

B. Depress the Man Runback pushbutton.

C. Depress 2A Emerg Stop A pushbutton.

D. Depress 2A VFD Lower Fast pushbutton.

Answer: C

K/A:
295014 INADVERTENT REACTIVITY ADDITION
AK1 Knowledge of the operational implications of the following concepts as they apply to INADVERTENT REACTIVITY ADDITION : (CFR: 41.8 to 41.10)
06 Abnormal reactivity additions

RO/SRO Rating: 3.8/3.9

Pedigree: New

Objective:
LOI-CLS-LP-302-C, Obj. 6 - List the Immediate Operator Actions required in accordance with AOP-03.0, Positive Reactivity Addition. *(LOCT)*

Reference: none

Cog Level: Fundamental Knowledge

Explanation:
The procedure directs using the Emerg Stop pushbutton thereby tripping the pump to remove the positive reactivity addition.

Distractor Analysis:

Choice A: Plausible because this is a 2A VFD control but is not directed to be used by the procedure.

Choice B: Plausible because this is a 2A VFD control that is directed to be used by the procedure to prevent a reactor scram.

Choice C: Correct Answer, see explanation

Choice D: Plausible because this is a 2A VFD control that is used normally to lower the speed of the pump but is not directed to be used by the procedure.

SRO Basis: N/A

4.1 Immediate Actions (continued)

3. **IF** reactor recirculation pump speed is rising,
THEN depress the affected pump(s) Emerg Stop pushbutton. .... ☐

44. 295015 1

Which one of the following completes the statements below IAW LEP-02, Alternate Control Rod Insertion?

The RWM is bypassed using a ___(1)___.

The reason that the RWM is bypassed is because the ___(2)___.

A. (1) keylock switch
   (2) Emergency Rod In Notch Override switch will not work when an Insert Block exists

B. (1) joystick
   (2) Emergency Rod In Notch Override switch will not work when an Insert Block exists

C. (1) keylock switch
   (2) Mode Switch in Shutdown generates a Control Rod Block

D. (1) joystick
   (2) Mode Switch in Shutdown generates a Control Rod Block

Answer: A

K/A:
295015 INCOMPLETE SCRAM
AK3 Knowledge of the reasons for the following responses as they apply to INCOMPLETE SCRAM: (CFR: 41.5 / 45.6)
01 Bypassing rod insertion blocks

RO/SRO Rating: 3.4/3.7

Pedigree: Bank, last used on the 2010-1 NRC Exam

Objective:
LOI-CLS-LP-007, Obj. 2d - State the purpose(s) of the following RWM components: Bypass Switch

Reference: None
Cog Level: Fundamental knowledge

Explanation:
LEP-02 is the procedure the we use to insert rods that have failed to insert on a scram. Direction is given to bypass the RWM which is accomplished by using a keylock switch in the RWM display console.
The RWM is bypassed to override the RWM Enforced Insert Block, allowing rods to be inserted using the Emergency Rod In Notch Override Switch.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because many other components are bypasses using a joystick controller.

Choice C: Plausible if examinee confuses the "Shutdown" withdraw block with an insert block.

Choice D: Plausible because many other components are bypasses using a joystick controller.
Plausible if examinee confuses the "Shutdown" withdraw block with an insert block.

SRO Basis: N/A

Step: Section 5

Source: PSTG RC/Q-6.2

Justification of Difference: Section 5 provides the plant-specific steps required to insert control rods with the Reactor Manual Control System (RMCS) defeating RWM interlocks. The plant-specific steps included in Section 5 are beyond the scope of the PSTG but are required to meet the intent of the PSTG.

Discussion: The purpose of Section 5 is to insert control rods with RMCS. This method is best applied when only a few control rods cannot be inserted, alternate methods are being performed which cannot be performed continuously, RPS cannot be reset, or individual control rod scrams are not effective. To assist in driving control rods it is possible to maximize drive pressure by starting both CRD pumps; throttling open Flow Control Valve, C11-F002A (F002B) [C12-F002A (F002B)]; and, if necessary, throttling closed Drive Pressure Valve, C11-PCV-F003 (C12-PCV-F003). Placing the RWM NORMAL/BYPASS switch to "BYPASS" to insert rods defeats the RWM interlocks.

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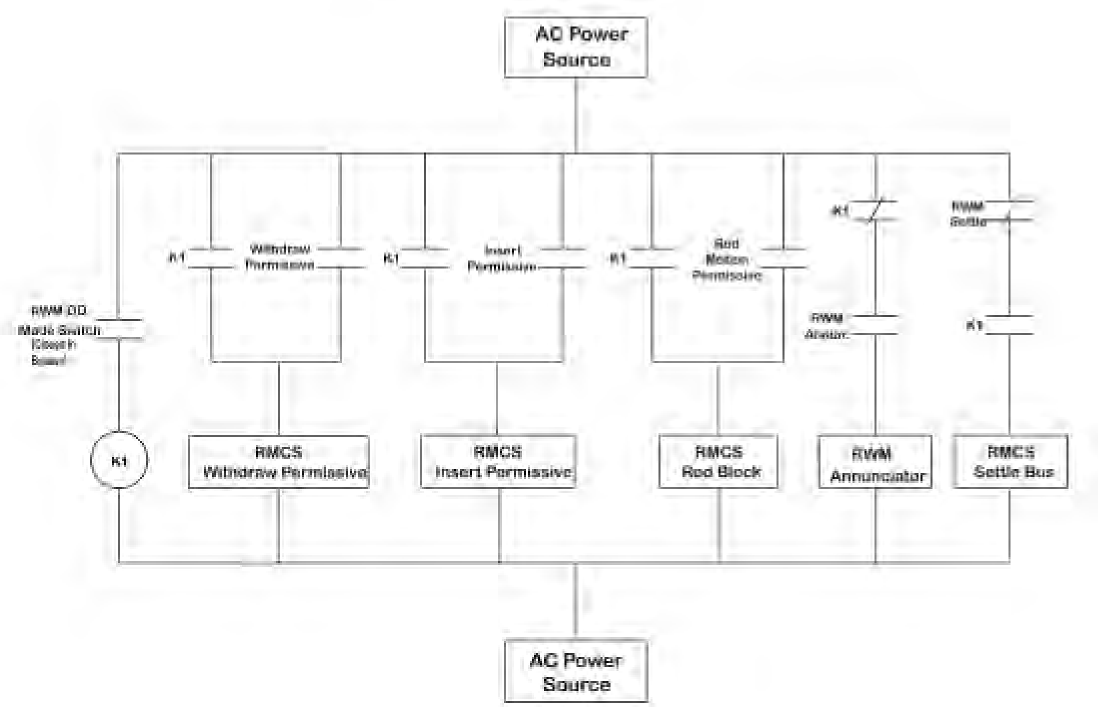
FIGURE 07.1-22
RWM-OD Bypass Function



3.4.4 RWM Operator Display Interface

The RWM-OD is interfaced to the RWM-CD to provide a system bypass capability. A contact set provides the bypass capability which includes (Figure 07.1-22):

- Insert permissive bypass, closed in bypass.
- Withdraw permissive bypass, closed in bypass.
- Rod drive block bypass, closed in bypass.
- Settle bypass, open in bypass.
- Annunciator bypass, open in bypass.

3.5.3 RWM Bypass

Placing the RWM-OD keylock mode switch in **BYPASS** will negate all RWM output contacts. If the RWM-CD keylock mode switch is in **OPER** (operate) and the RWM-OD keylock mode switch is in **BYPASS**, the RWM will continue to calculate, display, and enforce sequence conditions - however, the keylock switch contacts will prevent any actual rod blocks from occurring. Bypass capability continues to exist following loss of power to the RWM-OD.

The purpose of the RWM bypass capability is provided so that the RWM CD chassis can be removed, and replaced while the bypass switch is in the bypass state, without interrupting the system function.

When the RWM is bypassed, procedure 0GP-10 provides the only control rod movement constraints. Second operator verification of control rod select, position, and movement is employed using 0GP-11. Bypass is provided to perform maintenance and testing on the RWM without limiting plant operation. Bypass is also provided to enable control rods to be manually inserted without RWM restriction following a reactor scram.

3.6.4 Bypass - Chassis OPERATE Mode 4

The RWM-CD operates as in mode 1 to provide permissive and annunciation. The RWM-CD bypass switch overrides the RWM-CD outputs.

When the RWM is bypassed, the system provides insert and withdraw permissive information and no annunciation. The capability to receive and transmit rod position data, system status, and to receive and record rod scram time data is not inhibited.

3.6.5 Bypass - Chassis INOP Mode 5

When the RWM-OD is bypassed and the RWM-CD is in INOP, the only insert/withdrawal permissives are those provided by RMCS. There is no annunciation associated with this condition.

1.0 INTRODUCTION

1.1 System Purpose

The purpose of the Reactor Manual Control System (RMCS) is to allow the operator to control core reactivity by inserting and withdrawing control rods. The system consists of the electrical components and logic circuits required to monitor and manipulate the control rods. The Reactor Manual Control System also acts to block rod motion and/or selection in response to protective signals generated by other plant monitoring systems.

Supporting the RMCS is the Rod Position Information System (RPIS) which provides the operator with a means for determining the positions of all control rods in the core and for observing the position of a selected rod in relation to specific adjacent rods. RPIS also provides rod position and identification data to the process computer. For the purposes of this text, RPIS will be considered as a sub-system of RMCS.

3.1.3 Rod Motion Inhibits

Control rod movement can be inhibited by preventing rod selection, blocking rod withdrawal, or blocking rod insertion. These actions can be taken directly by various RMCS circuits or in response to signals generated by other plant monitoring systems.

Three conditions will prevent a control rod from being selected:

- RPIS inoperable
- Timer Malfunction Select Block
- Loss of 28 VDC to the select logic

A failure in the RPIS can prevent a rod from being selected or deselect a rod already selected. Failures that will cause the RPIS to be inoperative are:

- Master Clock Failure - the clock regulates the internal functions of the RPIS.
- Power Supply Failure - The RPIS uses power from the UPS which is converted to 24 VDC for use in the RPIS.
- Card removed or defective - Each position indicator provides information to an associated buffer card for processing and use in the RPIS.

ATTACHMENT 2
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0EOP-01-LEP-02

Alternate Control Rod Insertion

Step 5 provides the instructions needed to insert any control rods which are not fully inserted. This step bypasses the Rod Worth Minimizer and inserts the control rods with the Emergency Rod In Notch Override switch. This may be required if any control rods did not fully insert to position 00 or bounced back to position 02 on the reactor scram.

45. 295016 1

Which one of the following systems used during plant shutdown from outside the control room has both flow indication and flow control capability at the Remote Shutdown Panel?

A. CRD

B. RCIC

C. RHR Loop B

D. RHRSW Loop B

Answer: B

K/A:

295016 CONTROL ROOM ABANDONMENT

AK2 Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: (CFR: 41.7 / 45.8)

01 Remote shutdown panel

RO/SRO Rating: 4.4/4.5

Pedigree: Systems bank

Objective:

LOI-CLS-LP-062, Obj. 3 - List the systems that can be controlled from the Remote Shutdown Panel or local control stations.

Obj. 4 - List the plant parameters that can be monitored from the Remote Shutdown Panel.

Reference: None

Cog Level: Memory

Explanation:

The Remote Shutdown Panel has a flow indicating controller for RCIC that allows the operator to monitor and adjust RCIC system flow. RHR flow is indicated at the Remote Shutdown panel but adjusted by the MCC operator by throttling valves at the MCC while flow is monitored at the Remote Shutdown Panel. RHR SW is also adjusted at the MCC while monitoring RHR SW pump amps or NSW pump discharge pressure. CRD is operated from the associated switchgear

Distractor Analysis:

Choice A: Plausible because since CRD is operated during control room abandonment to augment level control and provide cooling to CRD mechanisms

Choice B: Correct Answer, see explanation

Choice C: Plausible because RHR is operated during control room abandonment for suppression pool cooling and shutdown cooling and has flow indication on the Remote Shutdown Panel

Choice D: Plausible because RHR SW must be operated during control room abandonment for support of the RHR system

SRO Basis: N/A

The equipment that may be operated at the Remote Shutdown Panel and locally to support a Shutdown from outside the Control Room include:

- EPA breakers for the RPS MG sets and Alternate Power Source permit shutting down the Reactor and closing the MSIVs if this action is not completed prior to evacuating the Control Room (located in Cable Spread Area).
- SRVs - Three SRVs (B,E,G) operated from the Remote Shutdown Panel to control Reactor Pressure while in Hot Shutdown and to cool down the Reactor.
- RCIC - can be started and secured locally in both the level control and pressure control modes; and controlled and monitored from the Remote Shutdown Panel. This is the primary means of controlling Reactor water level in Hot Shutdown and during the cooldown.
- CRD pumps - operated locally to provide cooling for the rod drives. A second pump may be started to assist in maintaining Reactor water level.
- Diesel Generators - Started locally and aligned to the E buses if power is loss to an E bus.
- RHR loop B - initially used for Suppression Pool cooling and then for Shutdown cooling when Reactor pressure is reduced to 50-100 psig. Operated locally and monitored at the Remote Shutdown Panel.
- RHR Service Water - operated locally to support RHR System operation.
- Nuclear Service Water - operated locally to support RHR System operation.
- Condensate System - Condensate Booster Pumps are tripped locally and the system is aligned to prevent injection to the Reactor vessel prior to Reactor pressure reaching 500 psig during the cooldown.
- HPCI - secured locally when no longer needed to maintain Reactor water level. If HPCI does not automatically initiate, it is not used to support the Shutdown.

Instrumentation provided at the Remote Shutdown Panel for plant parameters include:

• Reactor Water Level	1(2)-B21-LI-R604BX
• Reactor Water Level	1(2)-B21-LI-5977
• Reactor Water Level	1(2)-B21-LI-3331
• Reactor Pressure	1(2)-C32-PI-3332
• Drywell Pressure	1(2)-CAC-PI-3341
• Drywell Temperature	1(2)-CAC-TR-778 Pts. 1,3, and 4)
• Suppression Pool Level	1(2)-CAC-LI-3342
• Suppression Pool Temperature	1(2)-CAC-TR-778 (Pts. 5,6,7)
• RCIC System Flow	1(2)-E51-FIC-3325
• RHR System Loop B Flow	1(2)-E11-FI-3338

46. 295018 1

Unit Two is performing a reactor startup.

The following events occur prior to rolling the main turbine:

Bus 2C experiences a fault and trips
Unit Two NSW header ruptures in the Service Water Building
All Unit Two Service Water pumps supplying the NSW Header are manually tripped

Which one of the following identifies the status of the Diesel Generators and the cooling water supply?

A. ONLY DG4 is running with cooling water supplied from the Unit One NSW Pumps.

B. ONLY DG4 is running with cooling water supplied from the Unit Two CSW Pumps

C. DG2 is running with cooling water supplied from the Unit One NSW Pumps and DG4 is running with cooling water supplied from the Unit One NSW Pumps.

D. DG2 is running with cooling water supplied from the Unit One NSW Pumps and DG4 is running with cooling water supplied from the Unit Two CSW Pumps.

Answer: C

K/A:
295018 PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER
AA1 Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : (CFR: 41.7 / 45.6)
01 Backup systems

RO/SRO Rating: 3.3/3.4

Pedigree: mod from 2012 NRC exam (changed the unit, DGs and the BOP bus)

Objective:
AOI-CLS-LP-043, Obj. 6c - Discuss the automatic functions/interlocks associated with the Service Water System: Diesel Generator Cooling Water Supply Valves

Reference: None

Cog Level: High

Explanation: Divisional start signal will auto start both DG2 and 4. If service water pressure upstream of the jacket water heater exchanger remains below 5.6 psig for 30 seconds then the alternate unit supply valve (in this case Unit 1) will open and the normal supply valve will close. Since DG2 service water is normally from Unit 1, and Unit 1 service water system is intact, the normal supply from Unit 1 will remain in service. For DG4, the Unit 2 service water header is depressurized due to the rupture, therefore cooling water for DG 4 will align to the U1 NSW header.

Distractor Analysis:

Choice A: Plausible because if 2C 4160 deenergizes while UAT is energized (Unit online or in UAT backfeed), then only DG4 would start on loss of E-Bus voltage. U1 nuclear service water will automatically supply the diesel due to loss of U2 nuclear service water.

Choice B: Plausible because if 2C 4160 deenergizes while UAT is energized (Unit online or in UAT backfeed), then only DG4 would start on loss of E-Bus voltage. Without a casualty, conventional service water may be available to supply the nuclear header if aligned manually or aligned for auto start on the nuclear header.

Choice C: Correct answer, see explanation

Choice D: Plausible because UAT is deenergized during startup prior to synchronizing the generator to the grid. If 2C 4160 deenergizes while UAT is deenergized then a divisional DG start would result (DG 2 & 4).

SRO Basis: N/A

Question from 2012:

Unit One is performing a reactor startup.

The following events occur prior to rolling the main turbine:

Bus 1D experiences a fault and trips
Unit One NSW header ruptures in the Service Water Building

All Unit One Service Water pumps supplying the NSW Header are manually tripped IAW 0AOP-18.0, Nuclear Service Water System Failure.

The crew has completed all actions of 0AOP-18.0.

Which one of the following identifies the status of the Diesel Generator(s) and the cooling water supply?

A. ONLY DG1 is running with cooling water supplied from the Unit Two NSW Pumps.

B. ONLY DG1 is running with cooling water supplied from the Unit One CSW Pumps.

C. Both DGs 1 & 3 are running with cooling water supplied from the Unit Two NSW Pumps.

D. DG1 is running with cooling water supplied from the Unit One CSW Pumps and DG3 is running with cooling water supplied from the Unit Two NSW Pumps.

From SD-39.0, Emergency Diesel Generators

2.7 Diesel Generator Service Water (Figure 39-7)

Two service water supply lines provide service water to the tube side of each EDG set jacket water cooler. Each unit's Nuclear Service Water (NSW) System provides an independent source to all four Diesels. Diesel generator start and speed increase above 500 rpm opens the valve from the respective unit's NSW header. Should the service water pressure upstream of the jacket water heat exchanger remain below 5.6 psig for 30 seconds when the valve is open the alternate unit supply valve will open, then the normal supply valve will close. When the engine is shutdown and speed drops below 500 rpm the open valve will close. This switching sequence is initiated any time service water flow is lost when an EDG set is operating. Return flow of service water from all four jacket water coolers is routed to a common return line which discharges to SW Outfall Collection Tank via an 18" CPVC line.

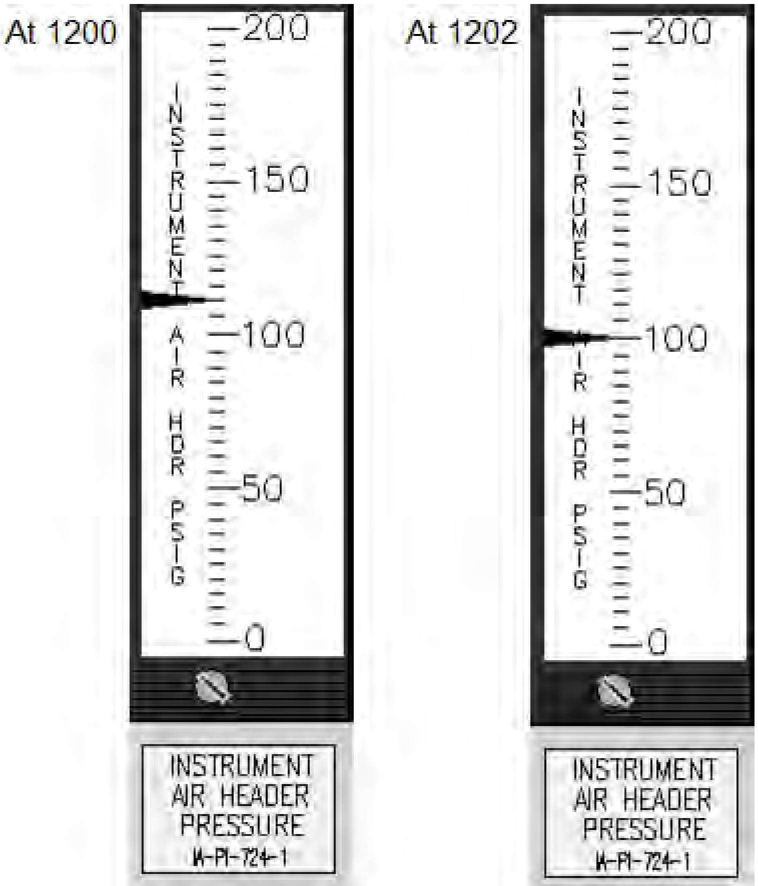
3. A EDG auto start signal will be generated for **EDGs 1 and 3 (2 and 4)** if any one of the following conditions exists (Figure 39-13):

- Loss of 1C or 2C 4160 BUS will cause EDGs. 2 & 4 to start
- Loss of 1D or 2D 4160V BUS will cause EDGs 1 & 3 to start

The loss of BOP bus is sensed by undervoltage on the secondary side of the UAT with the UAT to D(C) bus breaker open **OR** undervoltage on the secondary side of the SAT with the SAT to D(C) bus breaker open **AND** BOP bus undervoltage.

47. 295019 1

During operation at rated power with the instrument air NOT cross-tied, the following indication is observed:



Assuming the situation continues to degrade at the current rate, which one of the following represents the earliest time that the MSIVs may start drifting closed IAW 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures?

A. 1203

B. 1208

C. 1210

D. 1212

Answer: B

K/A:

295019 PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR

G2.04.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating: 4.2/4.2

Pedigree: New

Objective:
LOI-CLS-LP-046-A, Obj. 14 - Predict the effect that a loss or malfunction of the Pneumatic System would have on plant operation.

Reference: None

Cog Level: High

Explanation:
Based on the rate that the air pressure is dropping (5#/min) at 1203 pressure would be 95#, 1208 - 70#, 1210 - 60#, and 1212 - 50#. IAW the AOP the MSIVs may start drifting at 70#.

Distractor Analysis:

Choice A: Plausible because the procedure identifies that at 95# a scram would be inserted.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the procedure identifies that at 60# alot of valves begin to fail.

Choice D: Plausible because the procedure identifies that at 50# SA/IA crosstie begins to fail closed.

SRO Basis: N/A

On a loss of plant air, the following general plant response should be expected (without operator action).

- approx. 70 psig	MSIVs may start drifting closed after a sustained loss at this pressure (due to accumulators on the valve operators)
- approx. 60 psig	Loss of control of the condensate and feed system. Pump minimum flow valves start to drift open, feedwater level control valves drift open, hotwell level control valves drift open, heater drain pump discharge valves drift open, SULCV drifts closed.
- approx. 50 psig	SA/IA cross-tie valves begin to close.
- approx. 40 psig	Control rods start drifting in due to outlet scram valves opening.

3.1.1 **IF** any of the following conditions exist, **THEN MANUALLY SCRAM** the reactor **AND PERFORM** 1(2)EOP-01-RSP concurrently with this procedure:

- Unable to maintain at least one division noninterruptible instrument air pressure greater than 95 psig

48. 295021 1

Unit Two is in Cold Shutdown with both Reactor Recirculation pumps shutdown. Shutdown Cooling (SDC) has been established using RHR Loop B.

Which one of the following completes the statement below for a loss of shutdown cooling under the above conditions IAW 0AOP-15.0, Loss of Shutdown Cooling?

RPV Level must be raised to at least ___(1)___ inches to establish ___(2)____.

A. (1) 254
(2) natural circulation

B. (1) 254
(2) feed and bleed evolutions

C. (1) 200
(2) natural circulation

D. (1) 200
(2) feed and bleed evolutions

Answer: C

K/A:
295021 LOSS OF SHUTDOWN COOLING
AK1 Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : (CFR: 41.8 to 41.10)
02 Thermal stratification

RO/SRO Rating: 3.3/3.4

Pedigree: Previous, used on the 2010-2 exam.

Objective:
CLS-LP-302-L Obj. 5b, State the reason(s) for the following actions taken during a loss of Shutdown Cooling: b. Maintaining reactor water level >200 inches

Reference: None

Cog Level: Fundamental Knowledge

Explanation:
IAW AOP-15 level is raised to 200 - 220 inches to establish natural circulation. 254 inches is the level for establishing flow through the MSL to the SRVs back to the torus. Feed and bleed systems do not require a level increase IAW the procedure.

Distractor Analysis:

Choice A: Plausible because 254 inches is a level that is addressed in AOP-15 and natural circulation is the reason for raising level.

Choice B: Plausible because 254 inches is a level that is addressed in AOP-15 and feed and bleed is directed in this procedure but it does not require level to be raised.

Choice C: Correct Answer, see explanation

Choice D: Plausible because 200 inches is correct and feed and bleed is directed in this procedure but it does not require level to be raised.

SRO Basis: N/A

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4.2 Supplementary Actions (continued)

2. **IF** forced circulation has been lost,
AND natural circulation has **NOT** been established,
THEN ensure reactor vessel water level is being maintained between 200 inches and 220 inches as read on B21-LI-R605A(B) (RPV Water Level),
OR as directed by the Unit CRS based on plant conditions until forced circulation is restored... ☐

LOSS OF SHUTDOWN COOLING	0AOP-15.0
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4.2 Supplementary Actions (continued)

NOTE

Raising RPV water level slowly using CRD is preferred to reduce stresses induced in the RPV vessel and piping when injecting cold water. RHR and Core Spray may be used if necessary, but should be considered only after determining other methods are **NOT** effective. ☐

j. **Raise AND maintain** reactor water level greater than 254 inches. ☐

12. **IF** necessary to minimize reactor coolant temperature rise,
THEN perform one of the following feed and bleed combinations: ....

<table>
<tr><th>FEED</th><th>BLEED</th></tr>
<tr><td>Cond/FW in accordance with:
1OP-32
2OP-32</td><td>RWCU Reject in accordance with:
1OP-14
2OP-14</td></tr>
<tr><td>CRD in accordance with:
1OP-08
2OP-08</td><td>Reactor Water Level Control using Main Steam Lines in accordance with:
1OP-32
2OP-32</td></tr>
<tr><td>Core Spray in accordance with:
1OP18
2OP-18</td><td rowspan="2">Maintaining RPV Level Using the Main Steam Line Drains with:
1OP-25
2OP-25</td></tr>
<tr><td>LPCI in accordance with:
1OP-17
2OP-17</td></tr>
</table>

49. 295023 1

Which one of the following is a Plant Design Feature credited for minimizing the radiological impact of a Design Bases Refueling Accident IAW the Updated Final Safety Analysis Report (UFSAR)?

A. Control Building Ventilation Radiation Monitoring System auto start of CREV.

B. Reactor Building Ventilation Radiation Monitoring System auto start of SBGT.

C. Refueling Bridge Boundary Zone Control System preventing fuel movements into forbidden areas.

D. Spent Fuel Pool and Cooling System maintaining spent fuel pool level greater than 23 feet over irradiated fuel.

Answer: B

K/A:
295023 REFUELING ACCIDENTS
AK2 Knowledge of the interrelations between REFUELING ACCIDENTS and the following: (CFR:41.7/45.8)
03 Radiation monitoring equipment

RO/SRO Rating: 3.4/3.6

Pedigree: Previous, last used on the 2010-2 exam.

Objective:
LOI-CLS-LP-109, Obj 9 - Decide which feature(s) of facility design act to mitigate the consequences of the following: a. Refueling Accident

Reference: None

Cog Level: Fundamental knowledge

Explanation:
Plant design features credited during a RFA are the reactor building radiation monitors provide detection and isolation AND SBGT provide venting and filtering.

Distractor Analysis:

Choice A: Plausible because the dose consequence calculation for the fuel handling accident does not credit automatic start of CREVS, however, it does assume that CREVS is manually initiated within 20 minutes of a dropped/damaged fuel assembly.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is a true statement but is not a design feature credited during FHA.

Choice D: Plausible because this is an initial condition assumption for a FHA but not a design feature credited.

SRO Basis: N/A

The process liquid radiation monitors for the service water and liquid radwaste discharges possess radiation detection and monitoring characteristics sufficient to inform plant operations personnel whenever radiation levels in the discharges rise above pre-set limits.

All alarm trip circuits can be tested by using test signals or portable gamma sources.

11.5.2.5 Reactor Building Ventilation Radiation Monitoring System

The objective of the Reactor Building Ventilation Radiation Monitoring System is to indicate whenever abnormal amounts of radioactive material exist in the reactor building ventilation exhaust, and to effect appropriate action so that the release of radioactive material to the environs is controlled.

The Reactor Building Ventilation Radiation Monitoring System provides a clear indication to operations personnel whenever abnormal amounts of radioactivity exist in the reactor building ventilation exhaust.

The Reactor Building Ventilation Radiation Monitoring System initiates appropriate action to control the release of radioactive material to the environs when abnormal amounts of radioactive material exist in the reactor building ventilation exhaust.

The Reactor Building Radiation Monitoring System is shown on Figure 11-42 and specifications are given on Table 11-15. Two independent monitoring systems are used. One monitors gross gamma radiation and initiates reactor building isolation when a high radiation level is encountered; the other, a gaseous sample analyzer system, monitors, noble gases, annunciates an alarm if radioactive material concentrations exceed preset levels, and collects particulates and halogens on filters for effluent accountability. The first monitoring system consists of two independent channels. Each channel includes a Geiger-Muller-type detector with sensitivities of 0.01 mR/m to 100 mR/hr, and a combined indicator and trip unit. Both channels share a two-pen strip chart recorder. The equipment is located in the control room except the detectors which are located in the reactor building ventilation exhaust duct. Each indicator and trip unit has its own power supply.

Each channel of the dual channel gamma monitor has one trip and one alarm output. The upscale trip indicates high radiation and the downscale alarm indicates instrument trouble. If there is one upscale trip, the main Reactor Building Heating and Ventilation System is shut down, and the standby gas treatment system is started. In addition, the high radiation signal is sent to the Primary Containment and Reactor Vessel Isolation Control System to initiate closure of the various primary containment purge and exhaust paths.

The gaseous system analyzer consists of six isokinetic probes which take samples from the exhaust vent. The use of isokinetic probes in the air stream gives sufficient accuracy for measuring radiation releases. The full flow of the sample is directed to one of two parallel filter trains. Each filter train contains a stepped paper-type particulate filter for particulate collection and a carbon filter which collects halogens. The air then flows through a gas-counting chamber to detect noble gases. The monitor uses a four-point logarithmic chart recorder. The recording and control equipment is located in the electronic equipment room.

The environmental and power supply design conditions are given on Table 11-16. The gaseous system analyzer has one setpoint which causes annunciation in the control room when this setpoint is exceeded.

The physical location and monitoring characteristics of the reactor building ventilation radiation monitoring channels are adequate to provide detection capability for abnormal amounts of radioactivity in the reactor building ventilation and initiate isolation. The redundancy and arrangement of channels is sufficient to ensure that no single failure can prevent isolation when required. The upscale trips meet the design requirements of IEEE Standard 279-1971. During refueling operations the monitoring system acts as an engineered safety feature against the consequences of the refueling accident.

The monitors are installed so as to be readily accessible for inspection, calibration, and testing. The Reactor Building Ventilation Radiation Monitoring System and the response of the Reactor Building Heating and Ventilation System, and the Standby Gas Treatment System are routinely tested.

1.2 Design Basis

The Control Building Heating, Ventilation, and Air Conditioning System is designed to permit continuous occupancy of the Control Room (Habitability Envelope) under normal operating conditions and under the postulated design basis accidents including a complete rupture of the chlorine tank car throughout the life of the plant. The design also includes Auto-start of CREV due to a LOCA signal in less than or equal to two minutes. The dose consequence calculation for the design basis Fuel Handling Accident does not credit an operable secondary containment or automatic start of the Control Room Emergency Ventilation System. It assumes the Control Room Emergency Ventilation System will be placed in the Filtered Recirculation Mode by the Control Room Operators within twenty (20) minutes of occurrence of the Fuel Handling Accident to maintain the 30-day dose to the operators in the Control Room within the 10CFR50.67 limits. This system is also designed to maintain optimum atmospheric conditions within the various Control Building areas for the safety of plant personnel and equipment, and to prevent the accumulation of an explosive mixture of hydrogen gas released from the plant batteries.

The HVAC equipment, controls, and ductwork supports are designed to seismic Class I and are protected by tornado-proof construction. Portions of the system are also safety related. Redundant ventilating, air conditioning, and emergency filtering equipment are provided to ensure proper environmental conditions within the Control Room, computer rooms, and the electronic workrooms.

2.5 Reactor Building Ventilation Radiation Monitoring System (Figure 11-5)

The air passing through the Reactor Building ventilation exhaust fan room is continuously monitored for gamma radiation by the Reactor Building Ventilation Radiation Monitoring System. The system consists of two instrument channels, A and B, each of which contains a sensor and converter (D12-RE-N010A and B), an indicator and trip unit (D12-RM-K609A and B), and a shared two-pen recorder (D12-RR-R605).

The sensor and converter is a Geiger-Mueller type instrument which monitors the Reactor Building ventilation exhaust fan room and supplies an electrical signal, generated by incident gamma rays, to the indicator and trip unit.

The indicator and trip unit located in the radiation monitoring cabinet (H12-P606) in the electronic equipment room conditions the input signal to drive a panel mounted meter and a recorder. The amplified signal also drives a dual trip circuit which actuates annunciators and interrupts permissive interlocks to initiate shutdown and isolation of the Reactor Building Ventilation System, closure of the outboard primary containment purge and vent valves, and auto starts SBGT Systems A and B. During refueling operations, the monitoring system acts as an engineered safety feature against the consequences of the refueling accident.

1.3.1 Refueling Platform

- A Fanuc Programmable Logic Controller (PLC) implements Refueling Platform control functions. It can also serve as a Boundary Zone Controller (BZC) to prevent hoist movement into forbidden areas; however this function is normally bypassed.

- A Programmable Multiple Access Controller (PMAC) monitors left and right end truck positions and independently controls the bridge drive motors located on each end truck to maintain alignment. This is called a Dual Motor Drive (DMD) system. PMAC also controls Trolley motor movement.

4. For the 172 fuel rods with postulated damage, the activity release from the breached fuel clad is based on the following gap inventory fractions (i.e., fraction of total fuel rod activity contained in the gaps between fuel pellets):

Krypton 85	10%
Other Noble Gases	5%
Iodine 131	8%
Other Halogens	5%
Alkali Metals (Cesium, Rubidium)	12%

5. Of the radioiodine released from the damaged fuel rods, 99.85 percent of the released iodine is assumed in the form of elemental iodine and 0.15 percent of the released iodine is assumed to be in the organic species.
6. A 24-hour decay period is assumed prior to fuel removal and transport.
7. The GE14 fuel pin pressure is assumed to be less than 1200 psig.
8. The total number of effective full length fuel rods per fuel assembly is 87.33 and the total number of fuel assemblies in the core is 560.

15.7.1.3.2 Fission Product Transport from Fuel

The following assumptions, which are consistent with the guidance given in Reference 15-13, are used in the calculation of fission product activity transport from the fuel during a postulated refueling accident:

1. All gap activity in the damaged fuel rods is assumed to be instantaneously released.
2. Radionuclides considered include the xenons, kryptons, bromines, iodines, cesiums, and rubidiums.
3. All particulate radionuclides released (bromines, cesiums, and rubidiums) are assumed to be retained in the pool (i.e., infinite decontamination factor).
4. All noble gas radionuclides released are assumed to escape from the pool.
5. A minimum water depth of 23 feet is assumed over the reactor core.
6. The decontamination factor for organic iodine is assumed to be 1.
7. The overall iodine decontamination factor for the 23-foot water depth over the reactor core is 200. With an initial iodine species breakdown of 99.85 percent elemental and 0.15 percent organic, the iodine species fractions escaping the pool are calculated to be 57 percent elemental and 43 percent organic.
8. All radionuclide releases from the pool are conservatively assumed to be released directly to the environment over a 2-hour period with no credit taken for any dilution, holdup, or Engineered Safety Features (ESF) filtration within either the primary or secondary containments.

50. 295024 1

The following conditions exist on Unit Two:

Drywell pressure	2 psig
Drywell temperature	180°F
Reactor water level	95 inches
Reactor pressure	450 psig
Drywell Cooler Override Switch position	NORMAL

Which one of the following completes the statements below?

The Drywell Cooler fans ___(1)___ running.

The DW Lower Vent Dampers are in the ___(2)___ position.

A. (1) are
   (2) MIN

B. (1) are
   (2) MAX

C. (1) are NOT
   (2) MIN

D. (1) are NOT
   (2) MAX

Answer: B

K/A:
295024 HIGH DRYWELL PRESSURE
EA1 Ability to operate and/or monitor the following as they apply to HIGH DRYWELL PRESSURE:
(CFR: 41.7 / 45.6)
14 Drywell ventilation system

RO/SRO Rating: 3.4/3.5

Pedigree: New

Objective:
LOI-CLS-LP-004-A, Obj 5a - Describe the following as they relate to the Drywell Cooling System: Signals and setpoints that automatically trip the Drywell Coolers.

Reference: None

Cog Level: High

Explanation:
On a LOCA signal (LL3 or HI DW Press with Low reactor pressure) the cooling fans will trip. On low scram air header pressure the dampers realign to the MAX position.

Distractor Analysis:

Choice A: Plausible because no LOCA signal exist and the normal position of these dampers are in the MIN position. On a scram (low scram air header pressure these realign to the MAX position.

Choice B: Correct Answer, see explanation

Choice C: Plausible because if a LOCA signal existed then the fans would have tripped and the normal position of these dampers are in the MIN position. On a scram (low scram air header pressure these realign to the MAX position.

Choice D: Plausible because if a LOCA signal existed then the fans would have tripped and the dampers have realigned to the MAX position.

SRO Basis: N/A

The Drywell Lower Vent dampers can be positioned to either MIN or MAX position by a two-position control switch on Panel XU-3. Normal plant operating position for these dampers is the MIN position. Placing these dampers to MAX position during plant operation may produce extreme temperature excursions in the upper drywell regions. Low scram air header pressure will reposition these dampers to the MAX position and automatically start any idle drywell cooling fan selected for AUTO.

The drywell coolers receive a LOCA trip signal from the Core Spray initiation relays.

51. 295025 1

A loss of off-site power occurs on Unit Two with the following plant conditions:

Reactor water level	200 inches - stable
HPCI	In MAN in pressure control
RCIC	Tripped, ready for restart
Reactor pressure	1125 psig and rising

If reactor pressure is allowed to continue to rise, which one of the following identifies the reason the HPCI system will trip?

A. Turbine overspeed

B. High reactor water level

C. Steam Line High Flow

D. High turbine exhaust pressure

Answer: B

K/A:
295025 HIGH REACTOR PRESSURE
EK3 Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE: (CFR: 41.5 / 45.6)
03 HPCI operation

RO/SRO Rating: 3.8/3.8

Pedigree: New

Objective:
LOI-CLS-LP-019, Obj 3 - Given plant conditions, predict how the HPCI System will respond to the following events: m. High RPV water level

Reference: None

Cog Level: High

Explanation:
With reactor pressure rising an SRV will open which will cause about a 10 swell in level. With level at 200 inches a trip of HPCI will occur at about 206 inches. All of the other distractors are also trips but would not be caused by high reactor pressure.

Distractor Analysis:

Choice A: Plausible because with pressure rising the student may think that the HPCI turbine may overspeed.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is a trip of the system and exhaust pressure would be higher than normal.

Choice D: Plausible because this is a trip of the system and exhaust pressure would be higher than normal.

SRO Basis: N/A

Table 19-7 - HPCI Trips		
Signal	Setpoint	Tech Spec
Turbine Overspeed	4600 rpm • 450 rpm(110% of original rated speed - 4000 rpm)	N/A
Reactor High Water Level	206"	≤207"
HPCI Pump Low Suction Pressure	15 inches after 13 sec. time delay	N/A
Turbine High Exhaust Pressure	157.5 psig	N/A
HPCI System Isolation	See Table 19-8	See Table 19-8
Manual Trip	N/A	N/A
Low Steam Line Pressure*	115 psig	≥104 psig

Table 19-8 - HPCI (PCIS Group 4) Isolation Signals		
Signal	Setpoint	Tech Spec
HPCI Equipment Area Temperature High	165°F	≤175°F
HPCI Steam Line Area High Differential Temperature	47°F	≤50°F
HPCI Steam Line Tunnel High Ambient Temperature**	165°F/190°F	≤200°F
HPCI Steam Line Area High Temperature	165°F	≤200°F
HPCI Steam Line High Flow	220%	≤275%
HPCI Steam Line Flow Time Delay	5 sec	≥ 4, ≤ 12 sec
HPCI Steam Line Low Pressure	115 psig	≥104 psig
Turbine Exhaust Diaphragm High Pressure	7 psig	≤9 psig
Manual*	N/A	N/A

- Opening an SRV with no injection can cause approximately 10" of swell
- Closing an SRV with no injection can cause approximately 20" of shrink
- Opening an SRV for one minute at 1000 psig results in:
  - A pressure reduction of approximately 150 psig
  - Torus water temperature rise of approximately 2-3°F
- To maintain reactor level with sustained opening of an SRV, establish an injection rate of 1500 gpm to 2000 gpm (0.75 to 1.0 Mlbm/hr).
- When performing an emergency depressurization for conditions other than low reactor level, when reactor pressure lowers to allow injection from low pressure systems, allow injection flow rate to reach a minimum of 3000 gpm to 4000 gpm (1.5 to 2.0 Mlbm/hr) to prevent reactor level from lowering below TAF.
- HPCI/RCIC/CRD injection (4300/500/150 gpm) will sustain a reactor power level of approximately 20%.

52. 295026 1

Which one of the following completes the statements below?

The purpose of the RHR Heat Exchangers is to reject heat from the suppression pool to the ___(1)___ System.

In order to increase an established cooldown rate of the suppression pool IAW 2OP-17, Residual Heat Removal System Operating Procedure, throttle ___(2)___ E11-F048A, HX 2A Bypass Valve.

A. (1) Service Water
(2) open

B. (1) Service Water
(2) closed

C. (1) Reactor Building Closed Cooling Water
(2) open

D. (1) Reactor Building Closed Cooling Water
(2) closed

Answer: B

K/A:
295026 SUPPRESSION POOL HIGH WATER TEMPERATURE
G2.01.28 Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)

RO/SRO Rating: 4.1/4.1

Pedigree: New

Objective:
LOI-CLS-LP-017, Obj. 15 - Describe how the reactor cool down rate is controlled when the RHR system is in the Shutdown Cooling mode.

Reference: None

Cog Level: High

Explanation:
RHR Service water is the cooling medium for the RHR system. RBCCW does cool almost all other systems in the reactor building.
Throttling closed the bypass valve will provide more flow through the heat exchanger thereby providing more cooling. The student must know the flowpath to know this. If the bypass valve is on the cooling water side then this would provide a heatup not a cooldown.

Distractor Analysis:

Choice A: Plausible because SW is the cooling medium and if the valve is opened it provides the opposite answer from the desired answer thereby testing the students understanding of how the system works.

Choice B: Correct Answer, see explanation

Choice C: Plausible because RBCCW cools almost all components in the Rx Bldg and if the valve is opened it provides the opposite answer from the desired answer thereby testing the students understanding of how the system works.

Choice D: Plausible because RBCCW cools almost all components in the Rx Bldg and closed is correct.

SRO Basis: N/A

2.2 RHR Heat Exchangers

Two heat exchangers, one for Loop "A" and the other for Loop "B" are located in separate areas of the Reactor Building above the associated pumps at the 9' to 30' elevation. The heat exchangers are designed to remove the heat from the Suppression Pool water, the Reactor water, or, the Fuel Pool Cooling System water. Process water enters the carbon steel shell side and makes one pass through the heat exchanger, while RHR Service Water passes through copper-nickel "U" tubes to cool the water entering the shell side.

IF additional cooldown is desired, **THEN PERFORM** the following, as necessary, for each operating RHR loop while maintaining desired flow rate between 4500 and 10,000 gpm per loop:

a. **SLOWLY THROTTLE OPEN** *HX 2A(2B) OUTLET VLV, E11-F003A(B)*, as necessary.

b. **THROTTLE OPEN** *HX 2A(2B) SW DISCH VLV, E11-PDV-F068A(B)*, as necessary, to raise RHRSW flow rate.

c. **SLOWLY THROTTLE CLOSE** *HX 2A(2B) BYPASS VLV, E11-F048A(B)*, as necessary.

53. 295028 1

During an accident, reactor pressure and drywell reference leg area temperature are in the Unsafe region of the Reactor Saturation Limit.

Which one of the following reactor water level instruments are least likely to become unreliable due to reference leg flashing?

A. Fuel zone

B. Wide Range

C. Narrow Range

D. Shutdown Range

Answer: B

K/A:

295028 HIGH DRYWELL TEMPERATURE

EK2 Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: (CFR: 41.7 / 45.8)

03 Reactor water level indication

RO/SRO Rating: 3.6/3.8

Pedigree: Bank

Objective:

LOI-CLS-LP-001.2, Obj 5 - Explain the effect that the following will have on reactor vessel level and/or pressure indications: d) Reference/variable leg flashing ***(LOCT)***

Reference: none

Cog Level: high

Explanation:

During emergency conditions the Wide Range water level instruments may be used to determine reactor water level. However, Emergency Operating Procedures Caution 1 must be referenced to determine operability. The major portion of the reference leg is located in the reactor building and therefore, reactor building temperature is used in determining operability of the instruments. The REACTOR SATURATION LIMIT GRAPH is applicable to the Wide Range level instruments because of the small amount of reference leg located within the Drywell. The instruments may still be considered operable, even in the UNSAFE region, if level is greater than 20 inches

Distractor Analysis:

Choice A: Incorrect because this instrument has a long reference leg vertical drop in the drywell and would be unreliable if the reference leg flashed due to elevated drywell temperature.

Choice B: Correct Answer, see explanation

Choice C: Incorrect because this instrument has a long reference leg vertical drop in the drywell and would be unreliable if the reference leg flashed due to elevated drywell temperature.

Choice D: Incorrect because this instrument has a long reference leg vertical drop in the drywell and would be unreliable if the reference leg flashed due to elevated drywell temperature.

SRO Basis: N/A

The Wide Range water level transmitters are calibrated for:

- Reactor Dome Pressure 1000 psig
- Reference Leg DW Temperature 200°F
- Variable Leg DW Temperature 135°F
- Reactor Building Temperature 90°F

During emergency conditions the Wide Range water level instruments may be used to determine reactor water level. However, Emergency Operating Procedures Caution 1 must be referenced to determine operability. The major portion of the reference leg is located in the reactor building and therefore, reactor building temperature is used in determining operability of the instruments. The REACTOR SATURATION LIMIT GRAPH is applicable to the Wide Range level instruments because of the small amount of reference leg located within the Drywell. The instruments may still be considered operable, even in the UNSAFE region, if level is greater than 20 inches

The Narrow Range water level transmitters are calibrated for:

- Reactor Pressure 1030 psig
- Drywell Temperature 158 °F
- Reactor Building Temperature 70 °F

During emergency conditions the Narrow Range water level instruments may be used to determine reactor water level. However, Emergency Operating Procedures Caution 1 must be referenced to determine operability. Caution 1 contains two graphs related to determining operability of the Narrow Range water level instruments:

- REACTOR SATURATION LIMIT

Plots Reactor Pressure verses Reference Leg Area Drywell Temperature. Operating in the UNSAFE region of the graph would indicate potential boiling in the reference leg causing the indication to be unreliable.

This graph is generic for all level instruments however, the individual instrument caution specifies the Reference Leg Area Drywell Temperature to be used.

- INDICATED LEVEL VS REFERENCE LEG AREA DW TEMP

Plots Reference Leg Temperature verses Indicated Level. Operating in the UNSAFE region of the graph would indicate that the reference leg temperature is significantly higher than the calibration conditions. Under extreme conditions the instrument falsely indicates an on-scale and steadily increasing water level even though the actual water level may be well below the elevation of the instrument variable leg tap.

The Shutdown Range water level transmitters are calibrated for:

• Reactor Pressure	0 psig
• Drywell Temperature	70 °F
• Reactor Building Temperature	70 °F

During emergency conditions the Shutdown Range water level instruments may be used to determine reactor water level. However, Emergency Operating Procedures Caution 1 must be referenced to determine operability. Caution 1 contains two graphs related to determining operability of the Shutdown Range water level instruments:

- REACTOR SATURATION LIMIT
- INDICATED LEVEL VS REFERENCE LEG AREA DW TEMP

Fuel Zone Reactor Water Level Indication

The Fuel Zone reactor water level indication is used by the control room operator to monitor water level inside the shroud during emergency conditions. Indicating range is from minus 150 inches to plus 150 inches. The level indication is derived from level transmitters B21-LT-N036 and N037. The level transmitters provide input to an indicator (B21-LI-R610) and a recorder (B21-LR-615) located on P601.

The reference legs of N036 and N037 transmitters are supplied by uncompensated reference chambers D004A and D004B respectively. The variable leg of the transmitters are connected to calibrated instrument taps of Reactor Recirculation Jet Pumps 5 and 15 respectively. Since the location of the variable leg taps will result in these instruments indicating off-scale high whenever there is flow through the Jet Pumps, the instruments are **not** used with Reactor Recirculation Pumps in service.

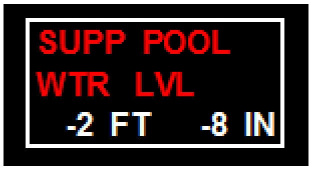
Fuel zone (accident) level instruments are calibrated to indicate most correctly when the reactor is depressurized and drywell temperature is elevated, with an absence of (no) Reactor Recirculation flow. Narrow and wide range (normal) level instruments are calibrated to read correctly at rated reactor pressures and normal reference leg and variable leg temperatures seen at 100 percent power.

54. 295030 1

With Unit One at rated power, the following control room indications are observed:

A-01 (3-7) *Suppression Chamber Lvl Hi/Lo* in alarm

ERFIS indication



Which one of the following completes the statements below?

The suppression chamber water level is ____(1)____.

This ____(2)____ require entry into an LCO 3.6.2.2, Suppression Pool Water Level, Action Statement.

A. (1) low
(2) does

B. (1) low
(2) does NOT

C. (1) high
(2) does

D. (1) high
(2) does NOT

Answer: A

K/A:
295030 LOW SUPPRESSION POOL WATER LEVEL
G2.02.38 Knowledge of conditions and limitations in the facility license. (CFR: 41.7 / 41.10 / 43.1 / 45.13)

RO/SRO Rating: 3.6/4.5

Pedigree: New

Objective:
LOI-CLS-LP-004.1, Obj 8 - Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM and COLR, determine whether given plant conditions meet minimum Technical Specifications requirements associated with the Secondary Containment system. (LOCT)

Reference: None

Cog Level: High

Explanation:
The alarm tells the operator the value is out of norm High or low. ERFIS indication turns red at -31 inches and would also turn red for a high level (-27 inches). converting the feet to inches indicates that this level is -32 inches which is low. the TS requires level to be within -27 to -31 inches, so this would require entry into the TS.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because this is a low alarm and the red indication is for EOP entry condition not TS entry.

Choice C: Plausible because this is a low not high alarm

Choice D: Plausible because this is a low not high alarm and the red indication is for EOP entry condition not TS entry.

SRO Basis: N/A

Suppression Pool Water Level
3.6.2.2

3.6 CONTAINMENT SYSTEMS

3.6.2.2 Suppression Pool Water Level

LCO 3.6.2.2 Suppression pool water level shall be ≥ -31 inches and ≤ -27 inches.

APPLICABILITY: MODES 1, 2, and 3.

Unit 1
APP A-01 3-7
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SUPPRESSION CHAMBER LVL HI/LO

AUTO ACTIONS

NONE

CAUSE

1. Suppression pool water level high (-27½ inches)
2. Suppression pool water level low (-30½ inches)
3. Circuit malfunction

122. SUPP POOL WTR LVL [Plant Status Matrix]

Event Status	Display Message	Color Code	Condition
Inactive/ Safe	(FEET AND INCHES)	Green	1. -2' 3½" (-27.5") > SP water level > -2' 6½" (-30.5")
Caution	(FEET AND INCHES)	Yellow	1. -2' 3" (-27") > level ≥ -2' 3½" (-27.5") - or - 2. -2' 6½" (-30.5") ≥ level > -2' 7" (-31")
Alarm	(FEET AND INCHES)	Red	1. -2' 3" (-27") ≤ level - or - 2. ≤ -2' 7" (-31")

55. 295031 1

During an accident, Unit Two plant conditions are:

Reactor water level	-35 inches, lowering
Reactor pressure	900 psig
Drywell average temp	185°F
Drywell ref leg temp	215°F
Injection sources	None available

Under these conditions, which one of the following is the LOWEST RPV water level that still assures adequate core cooling is being maintained?

(Reference provided)

A. -45 inches

B. -60 inches

C. -72.5 inches

D. -82.5 inches

Answer: C

K/A:

295031 REACTOR LOW WATER LEVEL

EA2 Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.10 / 43.5 / 45.13)

03 Reactor pressure

RO/SRO Rating: 4.2/4.2

Pedigree: New

Objective:

LOI-CLS-LP-300-G, Obj 12 - Given plant conditions and 0EOP-01-UG Reactor Water Level Caution (Caution 1), determine reactor water level relative to Low Level 4 (LL4) and Low Level 5 (LL5). (LOCT)

Reference: 0EOP-01-UG Attachment 6, Figures 17A, 18A & 19A.

Cog Level: hi

Explanation:

Since there is no correlation for determining reactor pressure from level indication, but there is from in opposite direction the chief examiner agreed that this question can be asked to meet the K/A.

Adequate core cooling exists per EOP-UG if RPV level is above LL5 with no injection, LL4 if there is injection to the reactor.

Distractor Analysis:

Choice A: Plausible because this is TAF with DW above 200°F.

Choice B: Plausible because this is LL4 with DW above 200°F.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because this would be correct if no injection source was available with DW below 200°F.

SRO Basis: N/A

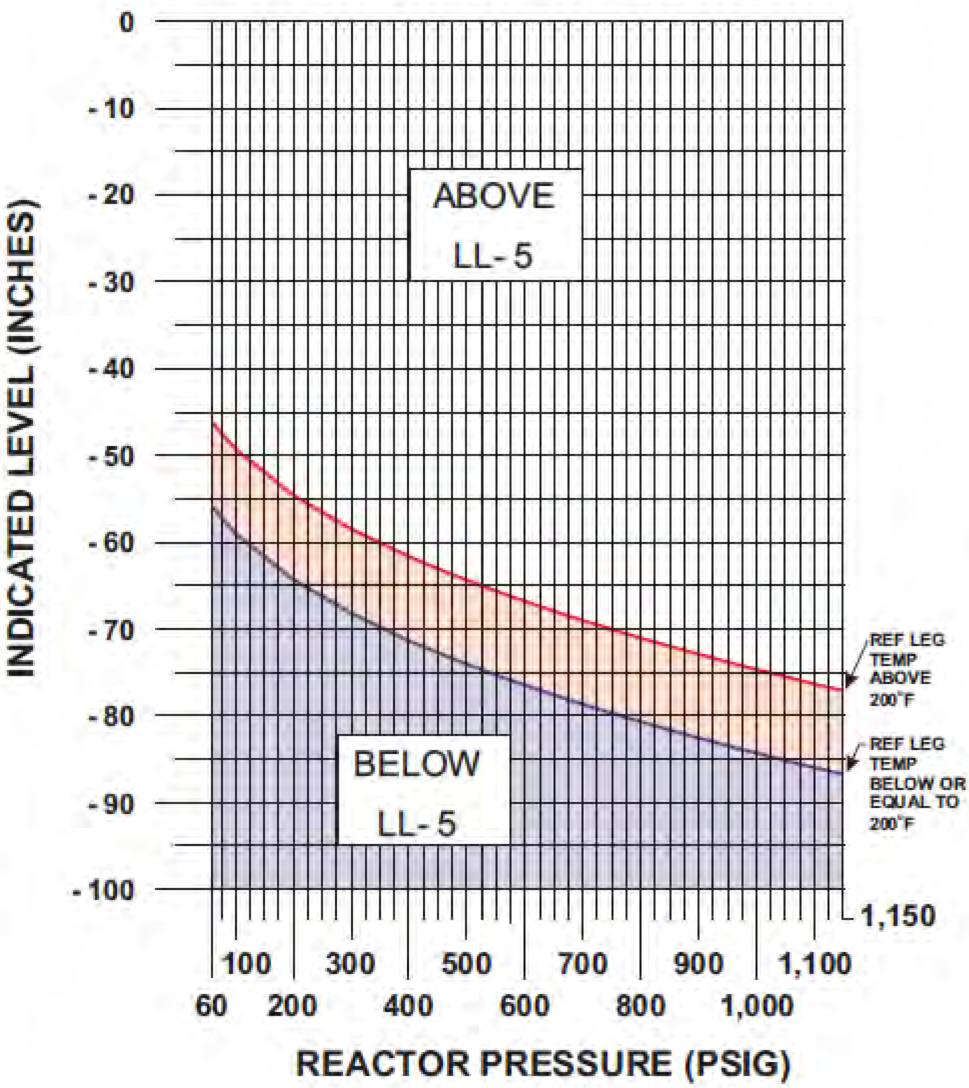
ADEQUATE CORE COOLING

Heat removal from the reactor sufficient to prevent rupturing the fuel clad.

Four viable mechanisms of adequate core cooling exist within the EOPs:

- Core submergence
- Steam cooling with injection of makeup water to the reactor
- Steam cooling without injection of makeup water to the reactor
- Reactor water level at jet pump suction with at least one core spray pump injecting into the reactor vessel at 5000 gpm.

UNIT 2
REACTOR WATER LEVEL AT LL- 5
DETERMINATION USING
B21- LI- R610 (N036) OR B21- LR- R615 (N037)



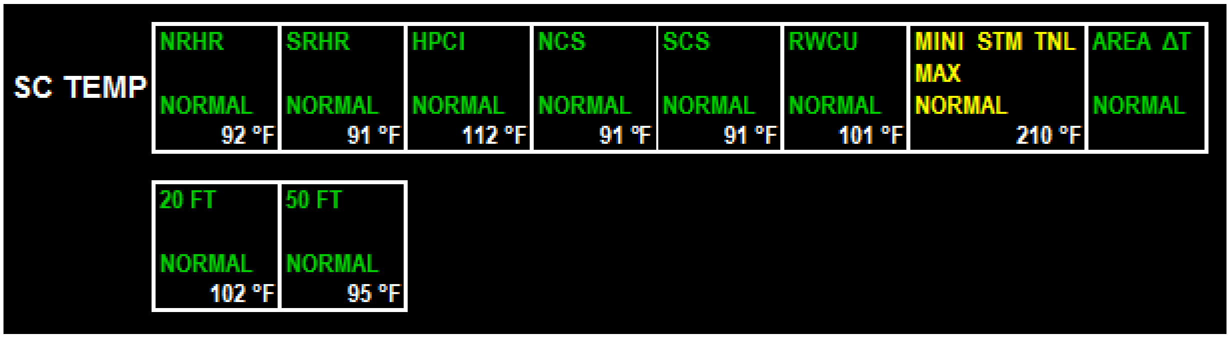
NOTE

WHEN REACTOR PRESSURE IS LESS THAN 60 PSIG, USE INDICATED LEVEL. LL- 5 IS - 45.0 INCHES.

56. 295032 1

Unit Two is operating at rated power.

The following ERFIS indications are observed ten minutes into the event:



Which one of the following identifies the status of the HPCI and RCIC systems based on the conditions above?

A. HPCI ONLY is isolated.

B. RCIC ONLY is isolated.

C. Both HPCI and RCIC are isolated.

D. Neither HPCI nor RCIC are isolated.

Answer: A

K/A:

295032 HIGH SECONDARY CONTAINMENT AREA TEMPERATURE

EK2 Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: (CFR: 41.7 / 45.8)

04 PCIS/NSSSS

RO/SRO Rating: 3.6/3.8

Pedigree: New

Objective:

LOI-CLS-LP-012, Obj 6 - Given plant conditions, determine if a Group Isolation should occur. ***(LOCT)***

Reference: None

Cog Level: high

Explanation:

Group 4 (HPCI) and 5 (RCIC) isolation on steam leak detection comes from Steam Leak Detection NUMAC modules (which provide area temperature monitoring, alarms and isolations). The leak is in the steam tunnel area (also known as mini steam tunnel, SPDS/ERFIS uses the term mini steam tunnel). Steam Tunnel area temperature channels of the NUMACs provide input to both group 4 and group 5 isolation logic circuits. The isolation setpoints are 165°F or 190°F dependent on instrument and the isolation setpoint has been exceeded. Since HPCI and RCIC steam lines both exit primary containment in the steam tunnel area, and to prevent simultaneous loss of HPCI and RCIC on a high steam tunnel area temperature, RCIC incorporates a 27 minute time delay on high steam line area temperature.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because the logic is the same for both systems with the exception of the time delay, so the student may get this concept backwards.

Choice C: Plausible because the group 4 and 5 isolation logic signal setpoints have been exceeded with the exception of the time delay for RCIC.

Choice D: Plausible because the student may think that both systems have a time delay or that the isolation is on Max Safe conditions.

SRO Basis: N/A

4. HPCI Steamline Tunnel Ambient Temp-High, Setpoint 165/190°F (See Table 12-2)

Three temperature switches (two in 'A' Logic, one in 'B' Logic) monitor HPCI steam tunnel area temperature and input to the NUMAC leak detection modules. One of the 'A' Logic switches is set for 165°F while the other two switches are set for 190°F.

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4. RCIC Steam Line Tunnel Ambient Temp. - High, Setpoint 165°/190°F (See Table 12-2)

Three temperature elements (two in trip system A and one in trip system B) monitor RCIC steam line area temperature and input to the NUMAC leak detection modules. One of the 'A' Logic switches is set for 165°F while the other two switches are set for 190°F. These isolation functions share the same NUMAC trip relay as the above temperatures. There is a ≈ 27 minute time delay for this isolation.

57. 295033 1

Which one of the following radiation annunciators requires entry into RRCP?

A. UA-03 (1-6) *RBCCW Liquid Process Rad High*

B. UA-03 (2-3) *Rx Bldg Roof Vent Rad High*

C. UA-03 (2-7) *Area Rad Rx Bldg High*

D. UA-03 (4-5) *Process Rx Bldg Vent Rad High*

Answer: B

K/A:

295033 HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS

G2.04.45 Ability to prioritize and interpret the significance of each annunciator or alarm.
(CFR: 41.10/43.5/45.3/45.12)

RO/SRO Rating: 4.1/4.3

Pedigree: Bank, Last used 2010-1 NRC Exam (Changed one distractor)

Objective:

LOI-CLS-LP-300-N Obj. 2 - Given plant conditions, determine if 0EOP-04-RRCP should be entered.

Reference: None

Cog Level: fundamental Knowledge

Explanation:

All of these alarms deal with rad conditions in secondary containment but only the Rx Bldg Roof Vent is an entry condition for RRCP.

Distractor Analysis:

Choice A: Plausible because this is a process system but the Service Water effluent is the entry condition not RCC.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is a precursor/effect of the other entry condition. (Blue bar alarm)

Choice D: Plausible because this annunciator provides indication of Secondary Containment abnormal rad conditon and is easily confused with the roof vent alarm.

SRO Basis: N/A

ENTRY CONDITIONS:

* MAIN STEAM LINE RAD HI ANNUN SETPOINT EXCEEDED (UA- 23,2- 6)
* PROCESS OFF- GAS RAD HI ANNUN SETPOINT EXCEEDED (UA- 03,5- 2) (SJAE)
* RX BLDG ROOF VENT RAD HIGH ANNUN SETPOINT EXCEEDED (UA- 03,2- 3)
* TURB BLDG VENT RAD HIGH ANNUN SETPOINT EXCEEDED (UA- 03,3- 3)
* PROCESS OG VENT PIPE RAD HI ANNUN SETPOINT EXCEEDED (UA- 03,6- 4) (STACK)
* SERVICE WTR EFFLUENT RAD HIGH ANNUN SETPOINT EXCEEDED (UA- 03,5- 5)
* ANY UNMONITORED OFF- SITE RADIOACTIVITY RELEASE
* CALCULATED DOSE RATE LIMIT OF "NOBLE GAS INSTANTANEOUS RELEASE RATE DETERMINATION" (E&RC- 2020) EXCEEDED

RR- 2

58. 295034 1

Unit Two is at rated power when the following annunciators are received:

UA-03 (5-2) *Process Off-Gas Rad High*
UA-03 (4-5) *Process Rx Bldg Vent Rad High*
UA-03 (3-5) *Process Rx Bldg Vent Rad Hi-Hi*

Which one of the following identifies the automatic actions that should occur?

A. PASS sample valves close and AOG-HCV-102, AOG System Bypass Valve, shuts (if open).

B. Group 6 initiation and AOG-HCV-102, AOG System Bypass Valve, shuts (if open).

C. Process Off-Gas Timer initiation and SBGT initiation.

D. Group 6 isolation and SBGT initiation.

Answer: D

K/A:
295034 SECONDARY CONTAINMENT VENTILATION HIGH RADIATION
EK2 Knowledge of the interrelations between SECONDARY CONTAINMENT VENTILATION HIGH RADIATION and the following: (CFR: 41.7 / 45.8)
03 SBGT

RO/SRO Rating: 4.3/4.5

Pedigree: New

Objective:
LOI-CLS-LP-010, Obj. 4 - Given plant conditions determine if SBGTs should have initiated. (LOCT)

Reference: None

Cog Level: High

Explanation:
The only alarm that has an auto action in the ones given is (5-4) which will isolate Rx Bldg Vent, start SBGT, and cause a Grp 6 isolation. A Process Off-Gas Rad Hi-Hi will start the timer and close the 102.

Distractor Analysis:

Choice A: Plausible because PASS does isolate but Process Off-Gas Rad Hi-Hi (not the High) will close the 102.

Choice B: Plausible because a Group 6 does occur but Process Off-Gas Rad Hi-Hi (not the High) will close the 102.

Choice C: Plausible because Process Off-Gas Rad Hi-Hi (not the High) initiates the timer and SBGT does initiate.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

Unit 2
APP UA-03 3-5
Page 1 of 2

PROCESS RX BLDG VENT RAD HI-HI

AUTO ACTIONS

1. Reactor Building Ventilation System trips and isolates
2. Standby gas treatment trains start
3. **IF** open, **THEN** inboard and outboard primary containment purge and vent valves close
4. PASS sample valves to torus close

Unit 2
APP UA-03 4-5
Page 1 of 1

PROCESS RX BLDG VENT RAD HIGH

AUTO ACTIONS

NONE

Unit 2
APP UA-03 5-2
Page 1 of 2

PROCESS OFF-GAS RAD HIGH

AUTO ACTIONS

NONE

Unit 2
APP UA-03 4-2
Page 1 of 2

PROCESS OFF-GAS RAD HI - HI

AUTO ACTIONS

1. Process off-gas timer is initiated if both channels are affected
2. **WHEN** process off-gas timer has timed out (15 minutes), if open, the following valves close:
   - *AOG SYSTEM BYPASS VALVE, AOG-HCV-102*
   - *OFF-GAS FILTER HOUSE LOOP SEAL RESERVOIR DRAIN VALVE, 2-OG-SV-4907*

59. 295036 1

Following a complete loss of RBCCW, a manual reactor scram was inserted.
Following the scram, the Scram Discharge Volume ruptured.
Plant conditions are:

Drywell average temp	190°F
Drywell pressure	2.3 psig
Rx Bldg 20' south temp	195°F

UA-12 (1-4) *South RHR Rm Flood Lvl Hi-Hi* is in alarm
UA-12 (1-3) *South CS Rm Flood Lvl Hi-Hi* is in alarm

Which one of the following identifies the operator action required by SCCP?

A. Perform emergency depressurization.

B. Reset RPS to isolate the primary system discharge.

C. Commence a reactor cooldown not to exceed 100°F/hr.

D. Rapidly depressurize to the main condenser irrespective of cooldown rate.

Answer: A

K/A:
295036 SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL
EA1 Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL : (CFR: 41.7 / 45.6)
03 Radwaste

RO/SRO Rating: 2.8/3.0

Pedigree: Bank, last used 2008 NRC exam

Objective:
LOI-CLS-LP-300-M, Obj 8c - Given plant conditions and the Secondary Containment Control Procedure, determine if any of the following are required: Emergency Depressurization (LOCT)

Reference: None

Cog Level: high

Explanation:
With the inability of the sumps to keep up with leakage (Radwaste) and two areas above max safe (flood level hi-hi alarms) with a primary system discharge (SDV) requires emergency depressurization. One of the steps in SCCP states to isolate all systems discharging into the area but RPS cannot be reset due to high drywell pressure.

Distractor Analysis:

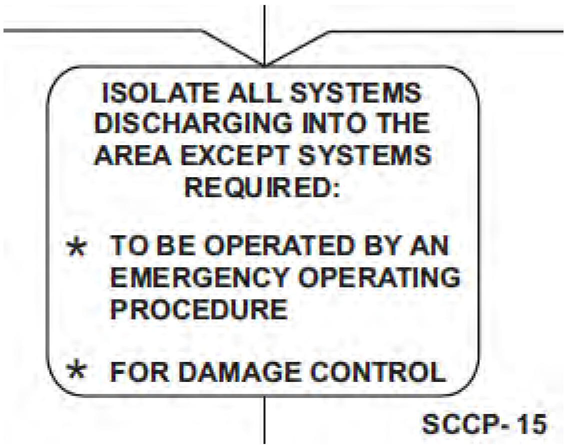
Choice A: Correct Answer, see explanation

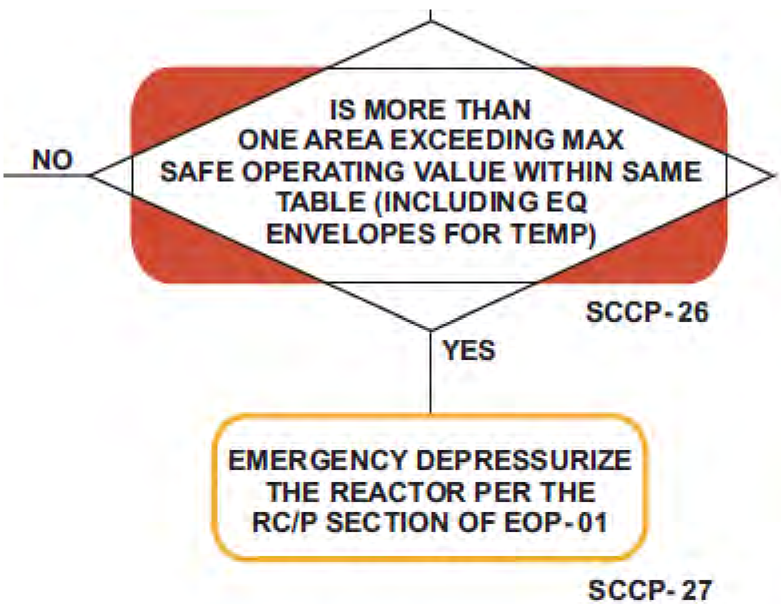
Choice B: Plausible because this would be an option to stop the leak but with the current conditions (hi DW press) this cannot be done.

Choice C: Plausible because this would be an option before the second area reach its max safe level.

Choice D: Plausible because this would be an option before the second area reach its max safe level or the student believes that the SDV is not a primary system.

SRO Basis: N/A





60. 295037 1

An ATWS condition currently exists on Unit Two with the following plant conditions:

Reactor Power	4%
Reactor pressure	controlled by EHC
Drywell pressure	2.1 psig
Reactor water level	95 inches
LEP-02 Section 3	jumpers have just been installed

Which one of the following completes the statements below concerning the required actions prior to resetting RPS IAW LEP-02, Alternate Control Rod Insertion, Section 3?

ARI is placed to ___(1)___ and then RESET.
The SDV Vents and Drains are confirmed to be ___(2)___.

A. (1) NORM
(2) open

B. (1) NORM
(2) closed

C. (1) INOP
(2) open

D. (1) INOP
(2) closed

Answer: D

K/A:
295037 SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN
EA1 Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: (CFR: 41.7 / 45.6)
01 Reactor Protection System

RO/SRO Rating: 4.6/4.6

Pedigree: New

Objective:
LOI-CLS-LP-300-J, Obj 6 - Given plant conditions and which steps have been completed, determine required operator actions in accordance with EOP-01-LEP-01,02, 03. (LOCT)

Reference: None

Cog Level: High

Explanation:
LEP-02 section 3 is for performing resetting of the scram and then re-scramming. After the jumpers are installed which bypass the scram signals, ARI is inhibitted, the current conditions have an ARI signal on LL2, so resetting would not work. The SDV V&D are confirmed to be closed prior to resetting the scram to make sure there is not an open pathway from the vessel when the scram is reset.

Distractor Analysis:

Choice A: Plausible because the ARI RESET switch is taken to reset, but the system is placed to INOP first. After the scram is reset then the SDV V&D are opened.

Choice B: Plausible because the ARI RESET switch is taken to reset, but the system is placed to INOP first.

Choice C: Plausible because after the scram is reset then the SDV V&D are opened

Choice D: Correct Answer, see explanation

SRO Basis: N/A

4. **INHIBIT** ARI by performing the following steps:

   a. **PLACE** *ARI AUTO/MANUAL INITIATION* switch, *C11(C12)-CS-5560*, to *INOP*. ☐

   b. **PLACE** *ARI RESET* switch (spring return), *C11(C12)-CS-5562*, to *RESET* **AND MAINTAIN** for a minimum of five (5) seconds, **THEN RELEASE**. ☐

   c. **CONFIRM** red *TRIP* light located above *ARI INITIATION, C11(C12)-CS-5561*, is off. ☐

5. **ENSURE** *DISCH VOL VENT & DRAIN TEST* switch is in *ISOLATE*. ☐

6. **CONFIRM** the following valves are closed:

   - *DISCH VOL VENT VLV C11(C12)-V139* ☐
   - *DISCH VOL VENT VLV C11(C12)-CV-F010* ☐
   - *DISCH VOL DRAIN VLV C11(C12)-V140* ☐
   - *DISCH VOL DRAIN VLV C11(C12)-CV-F011* ☐

7. **RESET** RPS. ☐

61. 295038 1

Unit Two has experienced a leak in the steam tunnel and the control building ventilation has realigned.

Which one of the following identifies:

(1) in what location will 1 mR/hr cause annunciator UA-03 (6-7) *Area Rad Control Room High* and

(2) the reason the control building ventilation has realigned?

A. (1) Control room.
   (2) To protect all Main Control Room personnel from elevated radiological conditions by processing intake air through the filter trains.

B. (1) Control room.
   (2) To protect personnel working in all areas of the control building from elevated radiological conditions by processing intake air through the filter trains.

C. (1) Ventilation intake duct.
   (2) To protect all Main Control Room personnel from elevated radiological conditions by processing intake air through the filter trains.

D. (1) Ventilation intake duct.
   (2) To protect personnel working in all areas of the control building from elevated radiological conditions by processing intake air through the filter trains.

Answer: A

K/A:

295038 HIGH OFF-SITE RELEASE RATE

EK3 Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.5 / 45.6)

03 Control room ventilation isolation

RO/SRO Rating: 3.7/3.9

Pedigree: New

Objective:

LOI-CLS-LP-037, Obj. 4 - Given plant conditions determine if signals exist that would cause the following to automatically start/open: (LOCT) a. Emergency Recirculation Fans

Reference: None

Cog Level: Hi

Explanation:

The CB HVAC isolates on 1mR in the control room or 7mR in the intake plenum and starts the CREV to filter the air for the control room only. The battery room fans continue to run but the cable spread and Mech Equip Room fans trip.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because 1 mR/hr in the control room is correct and even though the CBHVAC isolates the CREV starts to protect only the control room personnel not all control building personnel.

Choice C: Plausible because the ventilation intake does provide a an isolation signal at 7 mR/hr and the control room personnel is correct.

Choice D: Plausible because the ventilation intake does provide a an isolation signal but it is set for 7 mR/hr not 1 mR/hr and even though the CBHVAC isolates the CREV starts to protect only the control room personnel not all control building personnel.

SRO Basis: N/A

Unit 2
APP UA-03 6-7
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AREA RAD CONTROL ROOM HIGH

AUTO ACTIONS

CREV initiates

CAUSE

1. High radiation in one or more of the following areas of the Control Building:
   - Control Room (channel 1)
   - Unit 1 ventilation intake duct (channel 2)
   - Unit 2 ventilation intake duct (channel 3)

DEVICE/SETPOINTS

Channel 1 K2 Relay	1 mR/hr
Channel 2 K2 Relay	7 mR/hr
Channel 3 K2 Relay	7 mR/hr

1.1 System Purpose

The Control Building Heating, Ventilation and Air Conditioning (HVAC) System performs the following functions:

1.1.1 Maintains all occupied areas within the temperature ranges desired for human occupancy.

1.1.2 Maintains the various Control Building areas at the temperature conditions which provide for optimum operation of equipment.

1.1.3 Provides for isolation, the emergency filtration of Control Room air, and positive pressurization upon the detection of excessive radiation levels, smoke, or LOCA signal such that habitability conditions are maintained.

1.1.4 Provides for the automatic isolation of the Control Room atmosphere upon the detection of chlorine gas such that habitability conditions are maintained.

62. 300000 1

Unit Two is in MODE 3 following a seismic event with the following plant conditions:

Reactor level	55 inches
Reactor pressure	500 psig
Drywell pressure	9 psig

UA-01 (4-4) *Instr Air Press-Low* in Alarm
UA-01 (4-5) *Service Air Press-Low* in Alarm
UA-01 (1-2) *RB Inst Air Receiver 2B Press Low* in Alarm

Which one of the following completes the statements below?

RNA-SV-5481, Div II Backup N2 Rack Isol Vlv, is ___(1)___.

RNA-SV-5261, Div II Non-Inrpt RNA, is ___(2)___.

A. (1) open
(2) open

B. (1) open
(2) closed

C. (1) closed
(2) open

D. (1) closed
(2) closed

Answer: A

K/A:
300000 INSTRUMENT AIR SYSTEM
K3 Knowledge of the effect that a loss or malfunction of the INSTRUMENT AIR SYSTEM will have on the following: (CFR: 41.7 / 45.6)
01 Containment air system

RO/SRO Rating: 2.7/2.9

Pedigree: New

Objective:
LOI-CLS-LP-046-A, Obj. 7 - Given plant conditions, determine the effect(s) that the following conditions will have on the Pneumatic System: **(LOCT)** b. Low Instrument Air/Pneumatic Nitrogen (IAN/RNA/PNS) Header pressure

Reference: None

Cog Level: High

Explanation:
On a loss of pneumatics (<95#, as indicated by the UA-01 (1-2) alarm)) or a LOCA signal (<45 inches or >1.7 psig with <410#) the nitrogen Backup valves will open. The RNA isolation valve (5261) only closes on a LOCA signal.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because 5481 does open and the student may believe (common misconception) the RNA isolation also occurs

Choice C: Plausible because the LOCA siganl is not present so the student may believe the valves stay in thier normal position.

Choice D: Plausible because the student may get the logic backward for the iolstion valves.

SRO Basis: N/A

If the PNS/RNA header pressure drops to 95 psig, the Nitrogen Backup System valves will open to supply the Drywell loads, but the PNS/RNA isolation valves will not close. The common pneumatic loads will be supplied by the system that has the highest pressure. Check valves are installed to prevent the Nitrogen Backup System from supplying the other PNS/RNA loads.

63. 400000 1

Unit One was operating at rated power when a loss of the SAT occurs with the following plant conditions:

Reactor water level	120 inches
Reactor pressure	320 psig
Drywell pressure	13 psig
DG1	Running loaded
DG2	Tripped/Unavailable

Which one of the following completes the statements below concerning the operation of the RBCCW system?

A & C RBCCW pumps ___(1)___ running.

RCC-V-28 and RCC-V-52, DW Header Equipment Isolation Valves, ___(2)___.

A. (1) are
   (2) auto closed

B. (1) are
   (2) remain open

C. (1) are NOT
   (2) auto closed

D. (1) are NOT
   (2) remain open

Answer: D

K/A:
400000 COMPONENT COOLING WATER SYSTEM (CCWS)
A3 Ability to monitor automatic operations of the CCWS including: (CFR: 41.7 / 45.7)
01 Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS

RO/SRO Rating: 3.0/3.0

Pedigree: Bank, last used on 2007 NRC exam

Objective:
LOI-CLS-LP-021, Obj. 7 - List the signals that trip and lockout the RBCCW Pumps

Reference: None

Cog Level: High

Explanation:
A LOCA (hi DW with low reactor pressure) concurrent with Div I LOOP (primary L/O) trips A & C, no power for B (E2). RCC-V-28 and RCC-V-52 are primary containment isolation valves though they have no auto close feature. RBCCW HXs SW Inlet vlavles (SW-103/106) do auto close

Distractor Analysis:

Choice A: Plausible because if the student does not pick up on the LOCA signal this would be correct but the RCC valves do not auto close even though they are primary containment isolation valves. The RBCCW HXs SW Inlet vlavles (SW-103/106) do auto close though.

Choice B: Plausible because if the student does not pick up on the LOCA signal this would be correct.

Choice C: Plausible because there are no pumps running but the RCC valves do not outo close. The RBCCW HXs SW Inlet vlavles (SW-103/106) do auto close though.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

From SD-21

The A, B, and C pumps will automatically trip (regardless of control switch position) on a loss of off-site power concurrent with a loss of coolant accident. The D pump will shut down on loss of power and remain off until placed in OFF/RESET. This trip will reduce the load on the diesel generators during an accident condition with a loss of power.

2.8 Drywell Header Equipment Isolation Valves

The DW Header Equipment Isolation valves (RCC-V28 and RCC-V52) are primary containment isolation valves. Although they have no auto closure function, they allow for the manual isolation of all the components located in the drywell. The isolation valves are physically located on the 20' elevation of the Reactor Building. These isolation valves are powered from 480 VAC MCC 2(1)XC and controlled from control room panel XU-2.

64. 600000 1

A Unit One reactor building fire has occurred affecting safe shutdown Train B equipment.

Which one of the following identifies a component that is classified as ASSD Train B Equipment IAW 0ASSD-00, User's Guide?

A. CSW Pump 2C

B. NSW Pump 1B

C. HPCI System

D. RHR Pump 1A

Answer: B

K/A:
600000 PLANT FIRE ON SITE
AA2 Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE:
04 The fire's extent of potential operational damage to plant equipment

RO/SRO Rating: 2.8/3.1

Pedigree: New

Objective:
CLS-LP-304-09-4, Obj 5 - Given ASSD procedures, determine if a power source or equipment is classified as ASSD Train A or ASSD Train B. ***(LOCT)***
Reference: None

Cog Level: Fundamental Knowledge

Explanation:
All of the listed equipment is Train A with the exception of NSW Pump B.

Distractor Analysis:

Choice A: Plausible because CSW pump 2C is train A not Train B.

Choice B: Correct Answer, see explanation

Choice C: Plausible because HPCI is train A not Train B.

Choice D: Plausible because RHR Pump 1A is train A not Train B.

SRO Basis: N/A

3.3 ASSD Train A Equipment

3.3.5 HPCI System (Units 1 and 2)

3.3.7 Service Water System

1. Train A Service Water Pumps

   a. CSW Pump 2C (E1)

   b. CSW Pump 1B (E1)

   c. NSW Pump 2A (E3)

   d. CSW Pump 2A (E3)

   e. NSW Pump 1A (E1)

3.3.8 RHR Loop A (Units 1 and 2)

65. 700000 1

During rated power operation, plant status is:

UA-06 (1-2) *Gen Under Freq Relay* in alarm
Generator frequency is 59.2 Hertz

Which one of the following identifies why the turbine must be tripped if frequency remains at its present value?

To prevent damage to the:

A. Generator.

B. Main Transformer.

C. Low Pressure Turbine.

D. High Presure Turbine.

Answer: C

K/A:
700000 GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES
AK3 Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: (CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)
01 Reactor and turbine trip criteria

RO/SRO Rating: 3.9/4.2

Pedigree: Bank

Objective: none

Reference: None

Cog Level: High

Explanation:
Determines that frequency is low and is limited to five minutes of operation for this condition. A Caution from AOP-22, Grid Disturbances states the damage could occur in the low pressure blades.

Distractor Analysis:

Choice A: Plausible because system voltage will change but during grid disturbances.

Choice B: Plausible because system voltage will change but during grid disturbances low voltage conditions will accompany the low frequency condition.

Choice C: Correct Answer, see explanation

Choice D: Plausible because the damage is in the low pressure blades not the high pressure.

SRO Basis: N/A

CAUTION

- Off-frequency operation can stimulate resonance vibration in low pressure blades. .... ☐
- A total loss of off-site power (LOOP) should be anticipated if the turbine is tripped. .... ☐
- With grid voltage or frequency unstable or grid vulnerability identified, diesel generators should **<u>NOT</u>** be paralleled with any E bus connected to the grid since severe load swings may occur and possibly overload the diesel generators.................. ☐

5.0 GENERAL DISCUSSION

1. Typically, grid low voltage conditions will accompany grid low frequency. Emergency bus degraded voltage relays are designed to automatically operate and isolate the bus. Manual actions for bus isolation need only occur in response to protective equipment failures. Loads supplied from the electrical distribution system can typically operate over a wider frequency range without degradation. Given that operating frequency limitations of the grid and the main generators are far more restrictive than the allowable operating frequency range of loads, and that low frequency is typically accompanied by low voltage,

66. G2.01.25 1

A grid disturbance occurs with the following Unit One plant parameters:

Generator Load	980 MWe
Generator Reactive Load	160 MVARs, out
Generator Gas Pressure	50 psig

Which one of the following identifies the available options that will place the Unit within the Estimated Capability Curve?

(Reference provided)

A. Raise Gas Pressure or lower MWe.

B. Raise Gas Pressure or raise MVARs.

C. Raise Gas Pressure or lower MVARs.

D. Lower MWe or lower MVARs.

Answer: A

K/A:
G2.01.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.
(CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating: 3.9/4.2

Pedigree: Bank, last used on the 2008 NRC Exam

Objective:
CLS-LP-27, Obj. 9 - Given the Generator estimated capability curves, hydrogen pressure and either MVARS, MW, or power factor, determine the limit for MW and MVARS.

Reference: 1OP-27 Figure 1

Cog Level: High

Explanation:
Based on the conditions the student should plot the current location on the graph. Plot MWe along the bottom and MVARs up the side. Where these two points intersect, based on 50 psig gas pressure line is outside of the safe area. (Must be inside the curve to be safe) Lowering MWe or raising gas pressure are the only options. Lowering or raising MVARs would still be outside the curve.

Distractor Analysis:

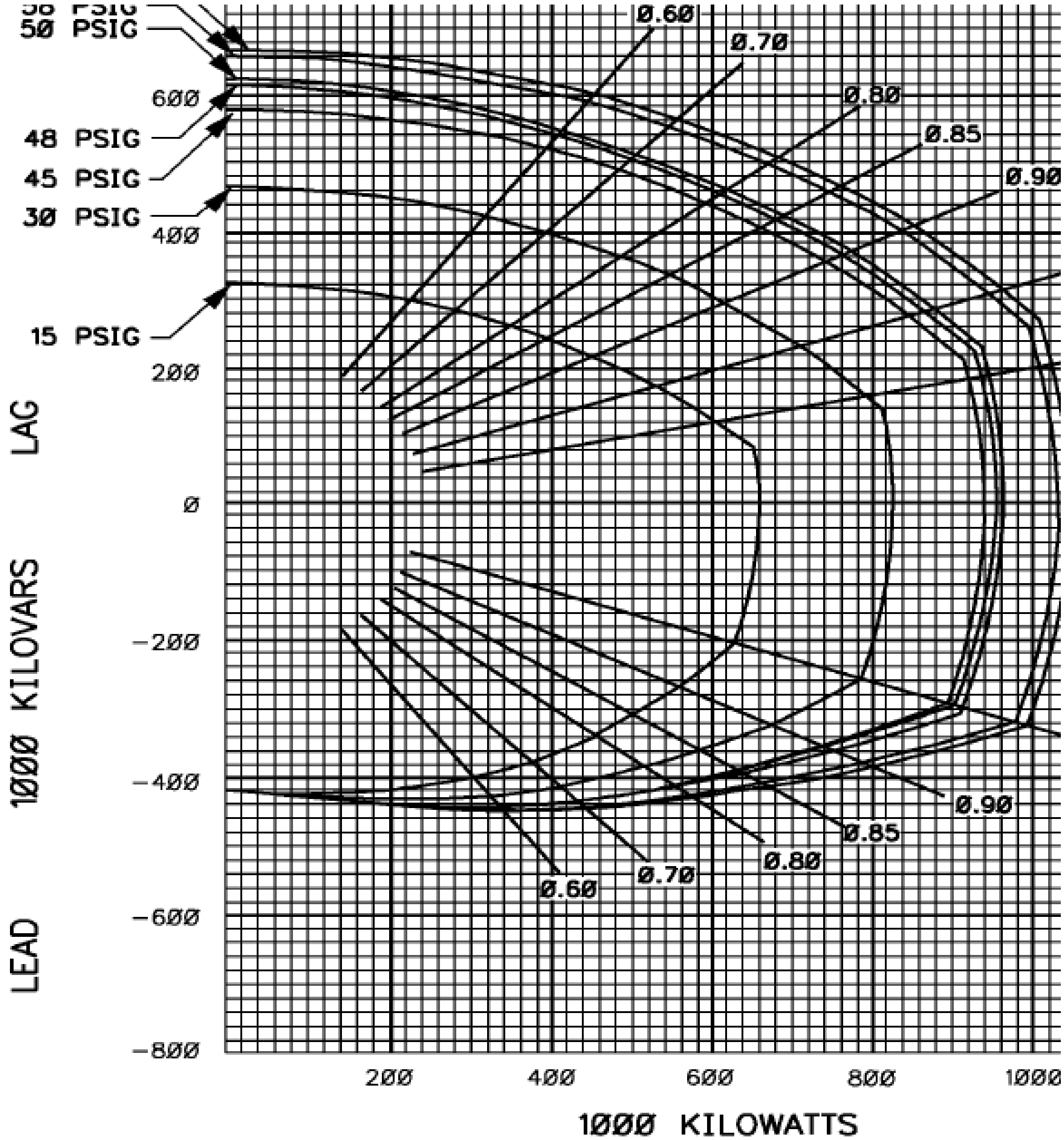
Choice A: Correct Answer, see explanation

Choice B: Plausible because see explanation

Choice C: Plausible because see explanation

Choice D: Plausible because see explanation

SRO Basis: N/A



58 PSIG
50 PSIG
48 PSIG
45 PSIG
30 PSIG
15 PSIG
LAG
LEAD
1000 KILOVARS
600
400
200
0
-200
-400
-600
-800
200
400
600
800
1000
1000 KILOWATTS
0.60
0.70
0.80
0.85
0.90

67. G2.01.27 1

Which one of the following completes the statement below concerning the purpose of the High Pressure Coolant Injection (HPCI) System IAW Technical Specifications Bases?

HPCI is designed to provide sufficient coolant injection to maintain the reactor core covered during a ___(1)___ Loss-Of-Coolant-Accident to maintain fuel cladding temperatures below ___(2)___.

A. (1) small break
   (2) 1800°F

B. (1) small break
   (2) 2200°F

C. (1) large break
   (2) 1800°F

D. (1) large break
   (2) 2200°F

Answer: B

K/A:
G2.01.27 Knowledge of system purpose and/or function (CFR: 41.7)

RO/SRO Rating: 3.9/4.0

Pedigree: Bank, last used on 2010-1 NRC Exam

Objective:
CLS-LP-019, Obj. 1 - State the purpose of the High Pressure Coolant Injection (HPCI) System.

Reference: None

Cog Level: Fundamental knowledge

Explanation:
The High Pressure Coolant Injection (HPCI) System was designed to provide sufficient coolant injection to maintain the Reactor core covered during a small line break Loss-Of-Coolant-Accident (LOCA) which does not result in rapid vessel depressurization, thus maintaining fuel cladding temperatures below 2200°F. The original design basis of the HPCI System was to provide part of the Emergency Core Cooling System (ECCS) function. HPCI system operation mitigated small break LOCAs where the depressurization function [Automatic Depressurization System (ADS) / SRVs] was assumed to fail.

Distractor Analysis:

Choice A: Plausible because small break is correct, and 1800°F is the number for if adequate core cooling can not be maintained by core submergence.

Choice B: Correct Answer, see explanation

Choice C: Plausible because HPCI is a high capacity, high pressure injection system which is easily mistaken for large break LOCA makeup requirements, and 1800°F is the number for if adequate core cooling can not be maintained by core submergence.

Choice D: Plausible because HPCI is a high capacity, high pressure injection system which is easily mistaken for large break LOCA makeup requirements, and 2200°F is the temperature that cladding will not exceed with core submergence.

SRO Basis: N/A

1.0 INTRODUCTION

1.1 System Purpose

The High Pressure Coolant Injection (HPCI) System was designed to provide sufficient coolant injection to maintain the Reactor core covered during a small line break Loss-Of-Coolant-Accident (LOCA) which does not result in rapid vessel depressurization, thus maintaining fuel cladding temperatures below 2200°F.

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MINIMUM ZERO-INJECTION REACTOR WATER LEVEL

The lowest reactor water level at which the covered portion of the reactor core will generate sufficient steam to preclude any clad temperature in the uncovered portion of the core from exceeding 1800°F. This water level is used by the Steam Cooling Procedure to preclude significant fuel damage and hydrogen generation for as long as possible (Unit 1 only: Figure 19; Unit 2 only: Figure 19A).

68. G2.01.31 1

TIP traces are in progress with all TIP drawer Mode Switches in Auto.

A small steam leak in containment causes drywell pressure to rise to 2.7 psig.

Which one of the following predicts the final TIP ball valve position indication(s) and also identifies all available location(s) for verifying their position?

A. Green light indication illuminated on each TIP drawer at Back Panel P607 ONLY.

B. White Valve Light illuminated on each TIP drawer at Back Panel P607 ONLY.

C. Red light indication illuminated on P601 Panel and on each TIP drawer at Back Panel P607.

D. Green light indication illuminated on P601 Panel and a white Valve Light illuminated on each TIP drawer at Back Panel P607.

Answer: D

K/A:

G2.01.31 Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup. (CFR: 41.10 / 45.12)

RO/SRO Rating: 4.6/4.3

Pedigree: Bank, last used on 2008 NRC Exam

Objective:

CLS-LP-09, Obj. 5b. explain the effects of the following on the TIP System: High Drywell Pressure

Reference: None

Cog Level: High

Explanation:

If drywell pressure reaches the PCIS Gp 2 isolation setpoint of 1.7 psig, TIP logic will initiate an automatic probe retract to the in-shield position and the TIP ball valves will auto close.

Indication of TIP ball valve position can be found on the P601 panel in the control room and the TIP back panel P607. The back panel indication white light is illuminated if the ball valve is closed (there is one on each drawer). The P601indication is red if any one of the 4 ball valves are open, and green if all 4 of the ball valves are closed.

Distractor Analysis:

Choice A: Plausible because a green light typically indicates a valve is closed, but the back panel lights are white.

Choice B: Plausible because these are illuminated but the P601 panel also has lights.

Choice C: Plausible because the red light would be on if no isolation had occurred.

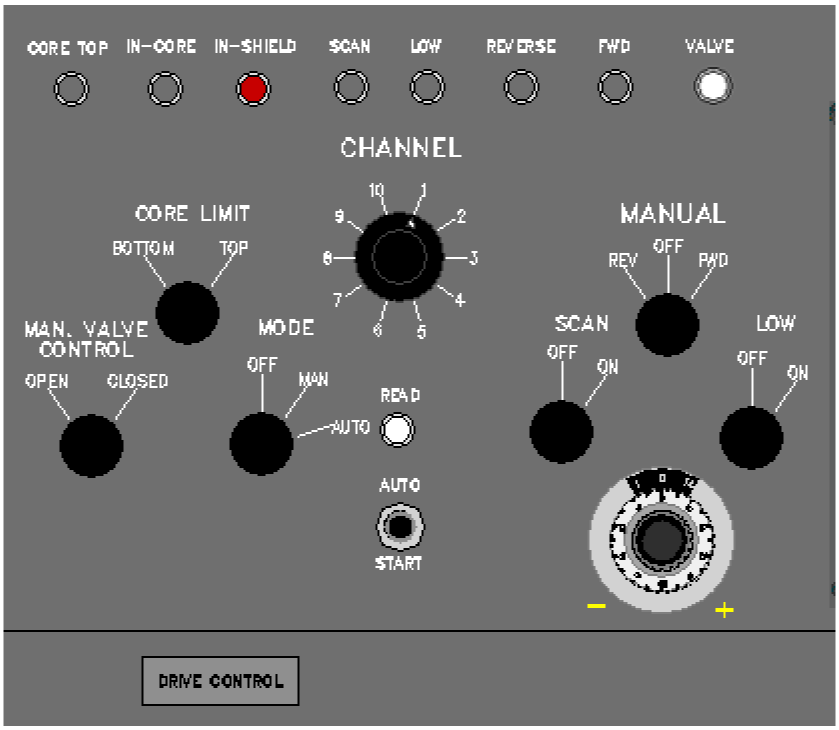
Choice D: Correct Answer, see explanation

SRO Basis: N/A

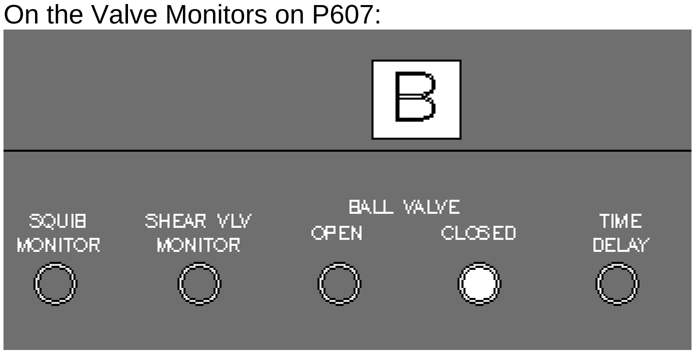
P601 indication:

(Note: the open indication is red)

On the Drive Control Unit on P607:



On the Valve Monitors on P607:



69. G2.02.03 1

Which one of the following Scram Immediate Operator actions has a different setpoint between Unit One and Unit Two?

A. Tripping of the main turbine.

B. Tripping of the first feed pump.

C. Master level controller setpoint setdown.

D. Placing the reactor mode switch to Shutdown.

Answer: D

K/A:

G2.02.03 (multi-unit license) Knowledge of the design, procedural, and operational differences between units. (CFR: 41.5 / 41.6 / 41.7 / 41.10 / 45.12)

RO/SRO Rating: 3.8/3.9

Pedigree: Last used on 08 NRC Exam

Objective:

CLS-LP-300-C, Obj. 2 - List the immediate operator actions for a reactor scram

Reference: None

Cog Level: Fundamental knowledge

Explanation:

The mode switch on Unit Two is not placed to shutdown until steam flow is less than 3 Mlbs/hr. This requirement does not exist on Unit One.

Distractor Analysis:

Choice A: Plausible because this is an operator immediate action but is not a design difference between the units.

Choice B: Plausible because this is an operator immediate action but is not a design difference between the units.

Choice C: Plausible because this is an operator immediate action but is not a design difference between the units.

Choice D: Correct Answer, see explanation

SRO Basis: N/A

3.3.1 Control Operator Immediate Actions

The reactor operator immediate actions are those actions which may be performed following a reactor scram prior to entering the scram procedure (EOP-01). These actions are not mandatory and shall not conflict with entering the scram procedure. All the reactor operator immediate actions are located in the scram procedure flowchart. There are no reactor operator immediate actions in EOP-02 through EOP-04. In the event the actions are not performed prior to entering the scram procedure, the scram procedure shall take precedence. The reactor operator immediate actions which should be memorized by reactor operators, are defined as follows:

1. **Unit 2 Only:** After steam flow is less than 3×10^6 lb/hr, PLACE the reactor mode switch to SHUTDOWN.

   Unit 1 Only: PLACE the reactor mode switch to SHUTDOWN.

2. **IF** reactor power is below 2% (APRM downscale trip), **THEN** TRIP the main turbine.

3. **ENSURE** the master reactor level controller setpoint is +170".

4. **IF** two reactor feed pumps are running, **AND** reactor vessel level is above +160" **AND** rising, **THEN** TRIP one.

70. G2.02.13 1

IAW OPS-NGGC-1301, Equipment Clearance, which one of the following identifies who can waive the requirement for a double valve isolation?

A. Assistant Operations Manager - Shift

B. Maintenance Manager

C. Work Week Manager

D. Plant Manager

Answer: A

K/A:
G2.02.13 Knowledge of tagging and clearance procedures. (CFR: 41.10 / 45.13)

RO/SRO Rating: 4.1/4.3

Pedigree: New

Objective:
OPI1301N, Obj. 3 - Determine the responsibilities of selected personnel for each clearance posistion.

Reference: None

Cog Level: Fundamental knowledge

Explanation:
The AOM - Shift is required to authorize this not using double isolation valves when required.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because Maintenance supervision is required to release a clearance.

Choice C: Plausible because the WWM is notified for changes to approved clearances.

Choice D: Plausible because Outage and Scheduling organization falls under the Plant Manager.

SRO Basis: N/A

From OPS-NGGC-1301:

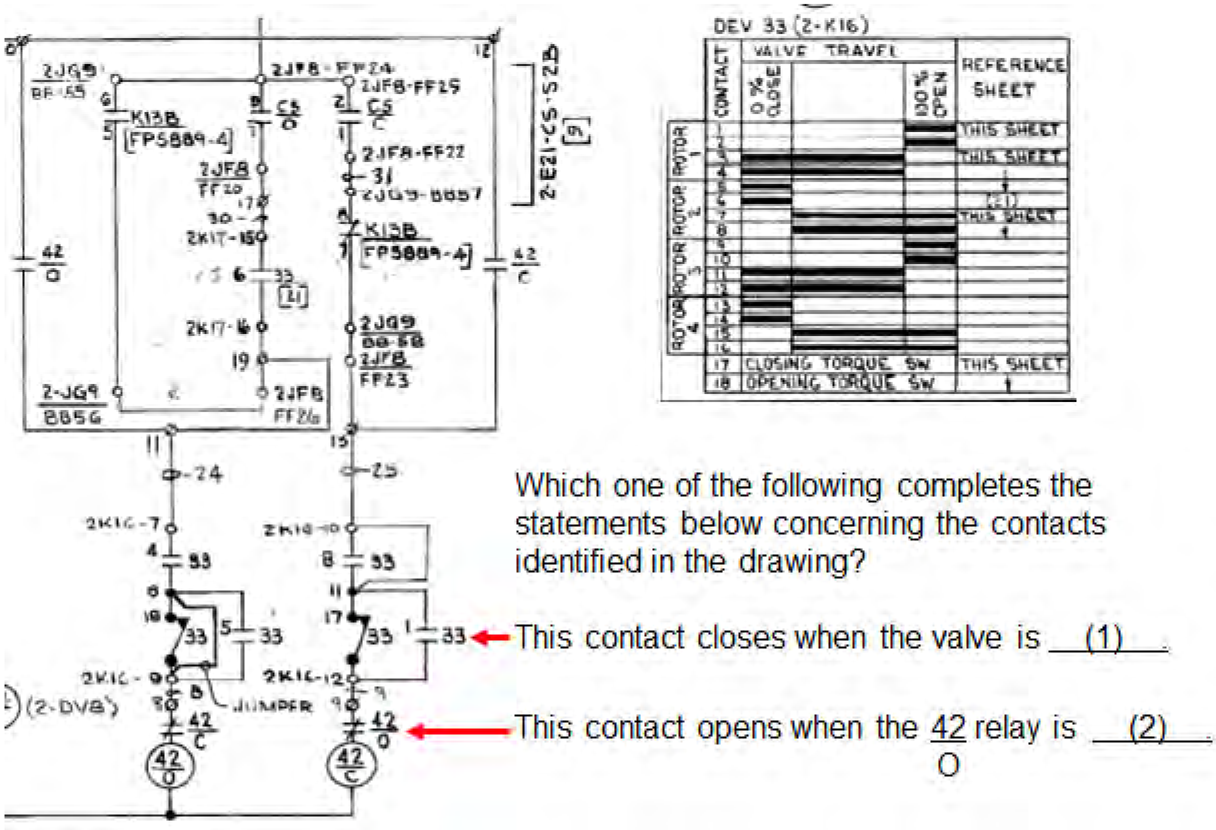
16. For systems that operate with temperatures greater than 200°F, pressures greater than 500 psig, caustic or acid systems (excluding boric acid):

  a. If plant design allows, then isolate from the work area by two in series closed valves when the system is to be breached.

  b. If plant design allows but double isolation is not used then AOM – Shift permission shall be obtained to hang any clearance that meets the above requirements and does not use double valve isolation. This permission and a notification to the workers of the clearance boundary limitations shall be noted in the Clearance Order Special Instructions.

  c. If plant design does not allow double valve isolation to meet the above requirements, Then AOM – Shift permission is not required, but a notification to the workers shall still be noted in the clearance Special Instructions.

71. G2.02.41 1



Which one of the following completes the statements below concerning the contacts identified in the drawing?

This contact closes when the valve is __(1)__.

This contact opens when the 42/O relay is __(2)__.

A. (1) Full open
   (2) energized

B. (1) Full open
   (2) de-energized

C. (1) NOT full open
   (2) energized

D. (1) NOT full open
   (2) de-energized

Answer: A

K/A:
G2.02.41 Ability to obtain and interpret station electrical and mechanical drawings.
(CFR: 41.10 / 45.12 / 45.13)

RO/SRO Rating: 3.5/3.9

Pedigree: New

Objective:
OPS-CLS-LP-111-A, Obj. 7 - Given a system wiring diagram, correctly interpret various logics.

Reference: None

Cog Level: Fundamental Knowledge

Explanation:
Cannot test on obtaining the controlled document so only wrote the question on interpreting the logic, this was agreed upon with the CE.
According to the switch development chart the 33 contact is closed for position 1 when the valve limit switches indicate the valve is in the full open position.   the 42/O contacts are b contacts and is open when this relay is energized.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because the first part is correct and this contact is a b contact meaning that it is closed when the relay is de-energized, instead of the usual closing when energized.

Choice C: Plausible because if the examinee thinks that the darken lines on the switch development table means that this is not where the contact is closed. Part 2 of this answer is correct.

Choice D: Plausible because   if the examinee thinks that the darken lines on the switch development table means that this is not where the contact is closed and this contact is a b contact meaning that it is closed when the relay is de-energized, instead of the usual closing when energized.

SRO Basis: N/A

72. G2.03.13 1

Which one of the following completes the following statements IAW 0OI-01.03, Non-Routine Activities, Section 5.6.1, Primary Containment Access.

The TIP system ___(1)___ required to be placed under clearance.

A clearance to prevent the withdrawal of control rods ___(2)___ required.

A. (1) is
   (2) is

B. (1) is
   (2) is not

C. (1) is not
   (2) is

D. (1) is not
   (2) is not

Answer: B

K/A:

G2.03.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.
(CFR: 41.12 / 43.4 / 45.9 / 45.10)

RO/SRO Rating: 3.4/3.8

Pedigree: New

Objective:
LOI-CLS-LP-201-D covers 0OI-01.03 but not specific to clearance requirements for DW entry at power.

Reference: None

Cog Level: fundamental knowledge

Explanation:
IAW OI-01.03 the TIP system is required to be under clearance and power can not be increased but no clearance is required.
Distractor Analysis:

Choice A: Plausible because a clearance on the TIP system is required and a power increase is not permitted but a clearance on rod withdrawal is not required.

Choice B: Correct Answer, see explanation

Choice C: Plausible because not all systems that affect rad conditions in the DW are placed under clearance and a power increase is not permitted but a clearance on rod withdrawal is not required.

Choice D: Plausible because not all systems that affect rad conditions in the DW are placed under clearance.

SRO Basis: N/A

CAUTION

TIP machine ball valves are Primary Containment Isolation Valves (PCIVs); Technical Specification Section 3.6.1.3 must be satisfied.

4. The TIPs shall be in the stored position (in shield) in the TIP room, and a clearance placed on each TIP ball valve (closed) and each TIP machine AUTO-MANUAL mode switch (OFF).

73. G2.03.14 1

Unit Two is in MODE 1 when the following alarms and indications occur:

UA-23 (2-6) *Main Steam Line Rad Hi*	In alarm
RWCU Conductivity Recorder	rising reactor water conductivity
Reactor power	remains steady
No other annunciators are in alarm	

Inititiation of which one of the following identifies the cause of these conditions?

A. Zinc injection.

B. Resin injection.

C. Hydrogen injection.

D. Noble metals injection.

Answer: B

K/A:

G2.03.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. (CFR: 41.12 / 43.4 / 45.10)

RO/SRO Rating: 3.4/3.8

Pedigree: New

Objective:

LOI-CLS-LP-014, Obj. 10 - Given plant conditions, predict how a failure of the RWCU System will affect the following: a. Reactor coolant conductivity. (LOCT)

Reference: None

Cog Level: Fundamental Knowledge

Explanation:

Resin, Hydrogen, and Noble metals injections all cause a rise in rad levels. Hydrogen injection will require the main steam line rad monitors to be adjusted based on the higher background readings. Noble metals injection will cause an initial increase and is performed after the unit ihas been running for greater than 100 days. Zinc injection is performed in to the feedpump suction but does not cause any increase in radiation. Radioactivity release procedure asks if it is a resin intrusion as indicated by rising conductivity.

Distractor Analysis:

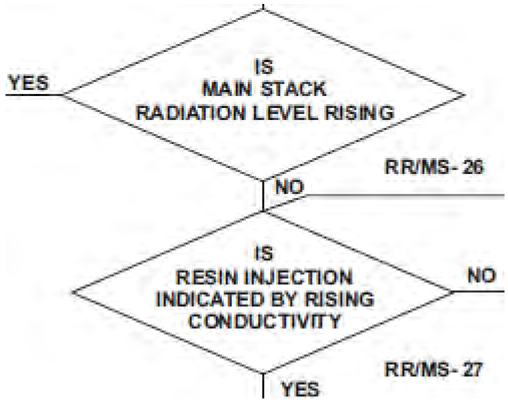
Choice A: Plausible because this is an injection into the feedwater system and takes up binding sites in the corrosion layer of the piping, leaving few locations for the Cobalt isotopes to plate out, thus lowering after shutdown rad levels.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is an injection into the feedwater system that would cause a rise in rad readings but would not cause the conductivity alarm.

Choice D: Plausible because this is an injection into the feedwater system that would initially cause a rise in rad levels but would not cause the conductivity alarm.

SRO Basis: N/A


YES
IS
MAIN STACK
RADIATION LEVEL RISING
RR/MS- 26
NO
IS
RESIN INJECTION
INDICATED BY RISING
CONDUCTIVITY
NO
RR/MS- 27
YES

74. G2.04.09 1

Alternate shutdown cooling using SRV's has been established IAW 0AOP-15.0, Loss of Shutdown Cooling. SRV B21-F013B is currently open. The cooldown rate is approaching 100°F/hr. The CRS has directed you to lower the cool down rate.

RHR A/C
B21-F013F B21-F013H
B21-F013G B21-F013J
B21-F013A B21-F013B B21-F013K
B21-F013C B21-F013D
B21-F013E B21-F013L

Which one of the following completes the statement below IAW the 0AOP-15.0 cooldown table above?

The RO can lower the cooldown rate by closing B21-F013B and opening _________.

A. B21-F013A

B. B21-F013C

C. B21-F013J

D. B21-F013K

Answer: B

K/A:
G2.04.09 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)

RO/SRO Rating: 3.8/4.2

Pedigree: New

Objective:
LOI-CLS-LP-302-L, Obj. 3 - Given plant conditions and AOP-15.0, Loss of Shutdown Cooling, determine the required supplementary actions.

Reference: None

Cog Level: High

Explanation:
IAW the table the top corresponds to the highest cooldown rate and the bottom of the table represent the lowest cooldown rate. SRV's within the same block produce a similar cooldown rate. To lower the cooldown rate (heatup) would require opening a SRV in a block lower than the current block, SRV C.

Distractor Analysis:

Choice A: Plausible because this SRV is in the table but it is within the same block as SRV B so no change in the cooldown rate would occur.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this SRV is in the table but it is a block above the cuurent one which would increase the cooldown not lower the cooldown.

Choice D: Plausible because this SRV is in the table but it is within the same block as SRV B so no change in the cooldown rate would occur.

SRO Basis: N/A

NOTE

All SRVs within the same block on the table will produce a similar cooldown rate; therefore, to effect a change in cooldown rate, an SRV in a different box must be used. ☐

	RHR A/C	RHR B/D	CS A	CS B
Highest Cooldown	B21-F013F B21-F013H	B21-F013A B21-F013B	B21-F013K	B21-F013E B21-F013L
	B21-F013G B21-F013J	B21-F013C B21-F013D	B21-F013G B21-F013J	B21-F013C B21-F013D
	B21-F013A B21-F013B B21-F013K	B21-F013E B21-F013F B21-F013H B21-F013L	B21-F013E B21-F013F B21-F013H B21-F013L	B21-F013A B21-F013B B21-F013K
	B21-F013C B21-F013D	B21-F013G B21-F013J	B21-F013C B21-F013D	B21-F013G B21-F013J
Lowest Cooldown	B21-F013E B21-F013L	B21-F013K	B21-F013A B21-F013B	B21-F013F B21-F013H

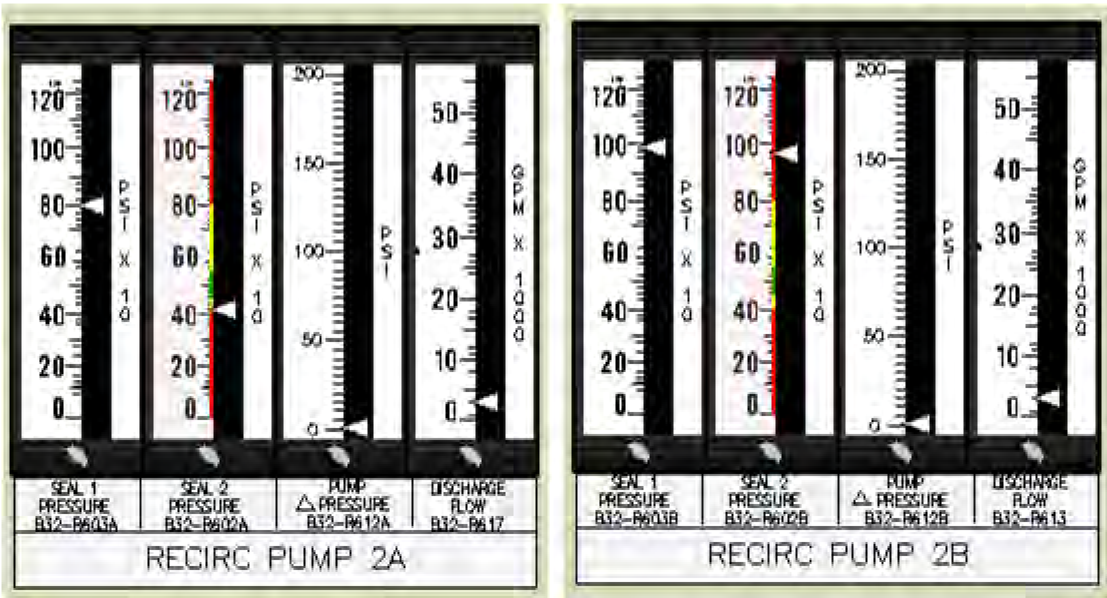
(5) __IF__ desired to adjust the cool down rate,
__THEN__:

(a) **Close** the open SRV ☐

(b) **Open** the next SRV that will adjust the cool down rate in the desired direction ☐

75. G2.04.21 1

Unit Two was at power when a trip and lockout of BOP Bus 2B required insertion of a manual reactor scram. Shortly after the scram, the following indications are noted:



Drywell pressure	1.4 psig, rising
Average drywell temp	140°F, rising

Which one of the following completes the statement below?

The crew will be required to enter ___(1)___ and isolate Recirc Pump ___(2)___.

A. (1) 0AOP-14.0
   (2) 2A

B. (1) 0AOP-14.0
   (2) 2B

C. (1) PCCP
   (2) 2A

D. (1) PCCP
   (2) 2B

Answer: A

K/A:

G2.04.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating: 4.0/4.6

Pedigree: New

Objective:
LOI-CLS-LP-302-D, Obj. 2 - Given plant conditions and AOP-14.0, determine the required supplementary actions.

Reference: None

Cog Level: High

Explanation:
Seal pressures should indicate approximately reactor pressure (both seals) following pump trip due to auto closure of the seal staging valve. Low seal pressures indicate seal failure. Either AOP-14 or PCCP will direct a recirc pump be isolated with indication of seal failure. Since drywell pressure is below 1.7 psig and drywell average temperature is below 150°F, no PCCP entry condition exists.

Distractor Analysis:

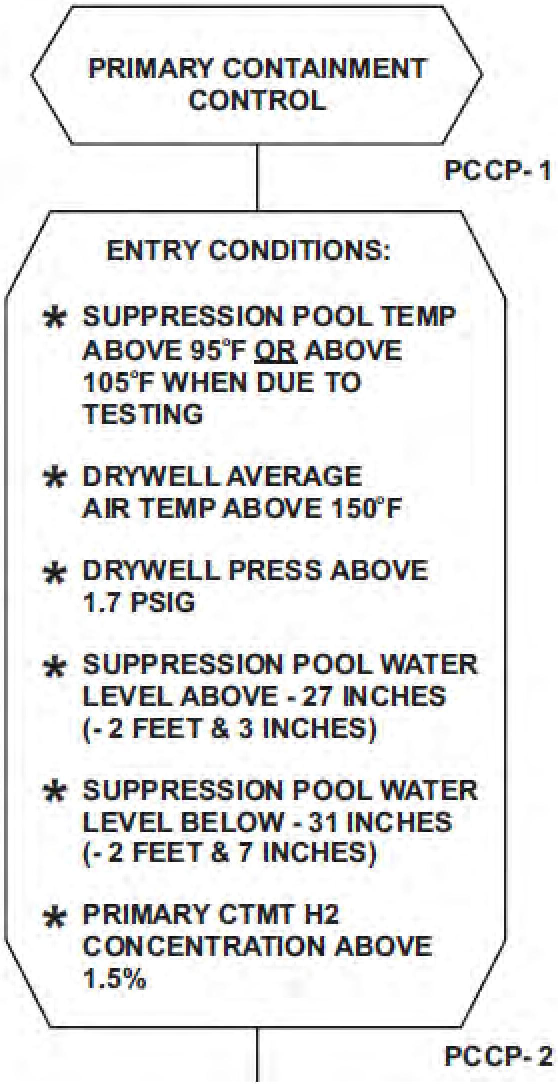
Choice A: Correct Answer, see explanation

Choice B: Plausible because AOP-14 is correct and this is the indications for a failed seal #1 on 2B RR pump

Choice C: Plausible because if the DW pressure or temperature was slightly higher than PCCP would be correct and this is the correct pump to trip.

Choice D: Plausible because if the DW pressure or temperature was slightly higher than PCCP would be correct and this is the indications for a failed seal #1 on 2B RR pump.

SRO Basis: N/A



The Seal Staging Valve is an air-operated valve which is interlocked with its associated Recirculation Pump such that the valve must be open in order to start the pump and will automatically close when the associated pump is secured. The Seal Staging Valve should be open at all times when the Recirculation Pump is operating to avoid subjecting the upper seal to full system pressure. Seal staging flow, normally less than 1.25 gpm, may be read locally in the Reactor Building on El. 20', across from the HCUs (Unit 2 only).