1. Given the following:

   - A reactor trip has occurred from 100% Rated Thermal Power.
   - E-0, "Reactor Trip or Safety Injection", is in progress at Step 2: Verify Turbine Trip.
   - HP turbine impulse pressure is 105 psig and lowering.
   - Turbine Stop Valve SV-1 CLOSED is LIT.
   - Turbine Stop Valve SV-2 CLOSED is LIT.
   - Turbine Stop Valve Closed Bistable light 44907-1107, Turbine Left Stop VLV Closed, is LIT.
   - Turbine Stop Valve Closed Bistable light 44907-1108, Turbine Right Stop VLV Closed, is NOT LIT.

In accordance with E-0, which of the following is the NEXT REQUIRED action?

A. Stop both EH oil pumps.

B. Manually trip the Turbine.

C. Check Bus 5 and Bus 6 energized.

D. Initiate a Main Steamline Isolation.

Answer: C

Cognitive Level:
Memory 1-P: The operator must possess knowledge of the procedure steps and cautions of E-0.

K/A:
007EA1.07 - Ability to operate and monitor the following as they apply to a reactor trip: MT/G trip; verification that the MT/G has been tripped. RO/SRO IMP 4.3/4.3

Objective:
RO4-04-LP002.009 - Given a set of plant conditions, RECOMMEND the appropriate procedural action to be taken while implementing E-0, Reactor Trip or Safety Injection.

Reference:
E-0, Reactor Trip or Safety Injection, step 2

Source:
New

Justification:

A INCORRECT This is also an RNO response if the stop valves are not closed. (see the justification for "C") However per the procedure the next step is to go to step 3.

B INCORRECT Manually trip the turbine is the next action ONLY if both the Stop valves are not closed. The operator only needs to check stop valve green lights - both lit OR turbine stop valves closed bistable lights both lit. This is a plausible distracter because one of the two indications for stop valves indicates that they are not closed. The RNO response to stop valves not closed is to manually trip the turbine.

C CORRECT Per E-0 the actions are complete for step 2 with no further action required. The next action required is to perform step 3, check 4160V Emergency AC Buses – BOTH ENERGIZED.

D INCORRECT Initiate a Main Steamline Isolation is a RNO action, only required if one of the two indications of stop valves closed is not present. This is a plausible distracter due to contradictory indications of the stop valve position.

2. Given the following:

    - The unit is in MODE 3.
    - PRZR Pressure Control Channel Selector switch is in the 2-3 position.
    - PRZR Level Control Channel Selector switch is in the 2-3 position.
    - Pressurizer pressure instruments:
        - PI-431 and PI-449 read off-scale low
        - PI-429 and PI-430 read 2185 psig and are lowering slowly
    - Pressurizer level instruments:
        - LI-428, Channel 3, is reading 75% and rapidly rising
        - LI-426 and LI-427 are reading 35% and are stable
    - Charging Pump 'A' is in automatic and speed is at minimum.
    - Annunciator 47043-J, Charging Pump In Auto High/Low Speed, is LIT.

What event is occurring?

A. PR-2A, PRZR PORV, has failed OPEN.

B. Pressurizer Pressure Master Controller failed High.

C. A LEAK has developed in the bellows at the upper tap for LT-428.

D. A Safety Injection has occurred due to the failure of PI-431 and PI-449 LOW.

Answer: C

Cognitive Level:

Analysis 3-SPK: Analyze the given conditions to determine what accident is occurring.

K/A:

008AA2.15 - Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: ESF control board, valve controls, and indicators. RO/SRO IMP 3.9/4.2

Objective:

RO2-05-LP36D.002 - DESCRIBE Pressurizer level control, include the following in the description:

1. System flow path (N/A)
2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   a. Level transmitters
   b. Level programmer
   c. Level recorder channel selector switch
   d. Level recorder
   e. Control room auto/manual control station
   f. Local auto/manual control station
3. Interfaces with the following plant systems:
   a. RCS
   b. CVCS
   c. Reactor protection

Reference:

OPERXK-100-10, Flow Diagram - Reactor Coolant System
XK-100-148, Logic Diagrams – Pressurizer Trip Signals
E-2038, Integrated Logic Diagram - Reactor Coolant System
E-2039, Integrated Logic Diagram - Reactor Coolant System

Source:

Bank - KNPP Initial Training Bank RO2-05-LP36D.002 001
Salem Exam - 6/98
Kewaunee Exam - 10/24/2000

Justification:

A INCORRECT PRZR PORV, PR-2A, has not failed open, the pressure instrument inputs to PR-2A, PI-431 and PI-449 have failed low.

B INCORRECT Pressurizer Pressure Master Controller has not failed high. PI-429 and PI-430 read 2185 psig and are dropping slowly due to a drop in level as a result of the failed level channel.

C CORRECT The event has to be a leak in the bellows at the upper tap for level channel 3 (LT-428). This is because LI-428 is reading 75% and rapidly rising while the other 2 channels LI-426 and LI-427 are reading 35% and are stable. This is basically a channel check situation.

D INCORRECT An SI does not occur due to not meeting SI coincidence of 2/3, PI-449 does not provide an input to Pressurizer pressure low SI.

3. Given the following:

 - ES-1.2, "Post LOCA Cooldown and Depressurization," is in progress following a small break LOCA.
 - SI pumps are injecting into the core.
 - RCS wide range temperature is 540ºF.
 - RCS subcooling is 56ºF.
 - Pressurizer heater control switches are in the OFF position.
 - Adverse Containment conditions are NOT present.
 - Pressurizer level is 0%.
 - RXCP B is running.
 - The Operator is directed to throttle open PS-1B, PRZR Spray Valve to refill the pressurizer.

How can the operator control PS-1B AND what is the reason for refilling the pressurizer?

	Control	Reason
A.	Manually Close PS-1A and then adjust the Pressurizer Spray Master Controller Automatic Setpoint	Establish Pressurizer level to allow automatic RCS pressure control
B.	Manually Close PS-1A and then adjust the Pressurizer Spray Master Controller in Manual	Raise RCS subcooling by causing the Safety Injection Accumulators to inject
C.	With PS-1B in Automatic adjust the Automatic Setpoint on the PS-1B controller	Prevent rapid RCS pressurization when Safety Injection flow exceeds break flow
D.	With PS-1B in Manual adjust the valve position of PS-1B	Add to the RCS inventory by reducing break flow and raising Safety Injection flow

Answer: D

Cognitive Level:

Comprehension 2-RI: The operator has to recognize the relationship between RCS pressure, SI flow and break flow for the given conditions.

K/A:

009G2.1.20 - Ability to interpret and execute procedure steps RO/SRO IMP 4.6/4.6

Objective:

RO-04-LP019.002 - Summarize purposes or bases of the following items as they relate to ES-1.2, "POST LOCA COOLDOWN AND DEPRESSURIZATION."

e. All Procedure Steps

RO4-04-SED08.003 - Perform a post LOCA cooldown and depressurization in accordance with ES-1.2, "POST LOCA COOLDOWN AND DEPRESSURIZATION"

References:

ES-1.2, Post LOCA Cooldown and Depressurization, Step 10

BKG-ES-1.2, Background Post LOCA Cooldown and Depressurization, KPS Step 10

Source:

New

Justification:

Using the RCS temperature and subcooling margin given, one can calculate the Przr (RCS) pressure. Przr temperature = RCS temperature + subcooling margin = 596ºF. Using Steam Tables this corresponds to a saturation pressure which is approximately 1482 psig. By reducing the Przr pressure while opening the spray valve, the SI flow will rise due to a higher ΔP between the SI Pump discharge pressure and RCS pressure, and RCS break flow will decrease with lower ΔP.

A INCORRECT With RCS pressure at saturation - The Pressurizer Master Controller in Automatic will not control. Lowest value is 1700 psig.

AND

Establishing PRZR level at this time for Automatic pressure control could be desirable, but is not required or the direct reason for establishing Pressurizer level by depressurizing the RCS.

B INCORRECT Controlling pressure with PS-1B in Automatic and the Pressurizer Master Controller in Manual could be done.

AND

With RCS pressure at 1482 psig, pressure is well above the pressure at which the SI Accumulators will inject (expected about 770 psig). Accumulators are designed for reflood for a large break LOCA. Lowing pressure to allow SI Accumulators to inject will result in loss/lowering of subcooling.

C INCORRECT There is no procedural direction to use PS-1B controller in automatic.

AND

With a small break LOCA SI flow and Break Flow will reach an equilibrium value based on RCS pressure. Overpressurization is not a concern.

D CORRECT Controlling depressurization using normal Pressurizer spray is procedurally driven, and since RXCP Pump 'B' is running, PS-1B would be the valve used, and since the Master Controller does not work below 1700 psig, manual is the method used.

AND

At this time the intent of refilling the Pressurizer by depressurizing the RCS is to reduce break flow and greater SI flow.

4. Given the following:

- The plant has experienced a Large Break LOCA.
- The crew has entered ECA-1.3, “Containment Sump Blockage”
- All attempts to establish containment sump recirculation have failed.
- Plant parameters and equipment status:

Parameter	Value	Trend
RWST Level	<3%	Stable
Boric Acid Storage Tanks	<9%	Stable
Core Exit Thermocouples	900ºF	Stable
RCS Wide Range Pressure	30 psig	Stable
Containment Pressure	5 psig	Slowly Lowering
Hottest Wide Range RCS Hot Leg	300ºF	Stable

Equipment	Status
SI Pumps	PULLOUT
RHR Pumps	PULLOUT
ICS	PULLOUT
RXCPs	PULLOUT
Charging Pumps ‘A’ & ‘B’	Running at Max (Suction from VCT)

- The crew has begun to DEPRESSURIZE All Intact Steam Generators to Atmospheric Pressure.
- The crew is waiting for concurrence from the Emergency Director to place the RHR system in service.
- The crew is maintaining RCS Heat Removal by continuing to dump steam from the intact SGs.

What is the basis for dumping steam from the intact SGs?

A. Maintain a differential temperature between RCS Hot and Cold legs to promote natural circulation.

B. Maintain proper thermal stratification in the SGs in the event of a SG Tube Rupture.

C. To ensure heat removal via Reflux Boiling/Natural Circulation.

D. To improve net positive suction to the RHR pumps.

Answer: C

Cognitive Level:

Comprehension 2-RI: The operator determines the interaction between SG and RCS during a Large Break LOCA and the basis for dumping steam

K/A:

011EK1.01 – Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCS: Natural circulation and cooling, including reflux boiling Emergency Procedures/Plan RO/SRO IMP 4.1/ 4.4

Objective:

RO4-05-LP005.006 – LIST the various Engineered Safety Features, which mitigate the consequences of DRCI events and IDENTIFY the conditions under which they would be actuated

Reference:

BKG-ECA 1.3, Background Containment Sump Blockage, Step 41
ECA-1.3, Containment Sump Blockage, KPS Step 41

Source:

RO&SRO Audit LXR Test Bank – 2008, and Kewaunee Exam – 4/18/2001

Justification:

A INCORRECT — With a large break LOCA heat removal is not by natural circulation. The Loops are not full. Heat removal is by reflux cooling.

B INCORRECT — Not a concern at this time, but does help equalize temperatures.

C CORRECT — Per the background document "If the RCS is not full of liquid at this time, it is especially important to keep the secondary system adequately full of water to promote reflux cooling."

D INCORRECT — NPSH for the RHR pumps is not the concern at this point. Maintain a cooling mechanism to prevent core damage is the concern.

5. Given the following:

   - The unit has been at 100% Rated Thermal Power for the last 42 days.
   - Annunciator 47013-I, RXCP A Seal Leakoff Flow High/Low, is LIT.
   - Annunciator 47015-I, RXCP A Standpipe High/Low, is LIT.
   - Reactor Coolant Pump 'A' #1 seal leakoff flow indicates 0.9 GPM.
   - Operators determine that Standpipe Level is HIGH.
   - Reactor Coolant Drain Tank (RCDT) level is rising faster than normal.

Which of the following has occurred?

A. #1 Seal has failed open

B. #2 Seal has failed open

C. #3 Seal blockage

D. #1 Seal blockage

Answer: B

Cognitive Level:

Comprehension 2-RI: Understand the interaction between seal injection, seal flow and seal failures and recognize the appropriate action.

K/A:

017AK2.10: Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions and the following: RCP indicators and controls.

Objective:

RO2-01-LP36A.004: DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Reactor Coolant Pump.

References:

AOP-RC-005, Abnormal Rxcp Operation, Section 2.3, Step 13.
ARP-47013-I, RXCP A Seal Leakoff Flow High/Low
ARP-47015-I, RXCP A Standpipe High/Low

Source:

KNPP Initial: RO2-01-LP36A.004 #54, Modified the stem and all four choices.

Justification:

AOP-RC-005, Symptoms and Entry Conditions, Step 2.3
No. 2 Seal malfunction:
RXCP A Seal Leakoff flow High/Low (47013-I)
RXCP Standpipe High/Low (47015-I)
Increasing level in RCDT

NOTE: From AOP-RC-005, page 15 of 25: "High Standpipe level indicates excessive #2 seal leakage. Low standpipe level indicates excessive #3 seal leakage.

A INCORRECT With the #1 Seal failed open, #1 seal flow would be out of specifications high.

B CORRECT See note above, for #2 Seal Failure indications.

C INCORRECT #3 Seal blockage would result in Standpipe level high but not decreased #1 seal leakoff flow.

D INCORRECT #1 Seal blocked would result in less #1 seal flow however, RCDT and Standpipe level would not rise.

6. Given the following:

   - The unit is at 100% Rated Thermal Power.
   - VCT level transmitter LT-112 (24015) fails high (100%).
   - Annunciator 47043-L, VCT Level High/Low, is LIT.
   - The operator is directed to perform manual makeups to the VCT as required to maintain VCT Level

Why does the operator take manual action to maintain VCT level?

To prevent . . .

A. the charging pumps from losing suction and cavitating because auto makeup will NOT maintain VCT level.

B. continuous automatic makeup caused by LD-27, VCT/Holdup Tank Divert Valve, shifting to DIVERT.

C. LD-14, LD Demin High Temp Divert Valve, from shifting to DIVERT caused by a rise in letdown flow.

D. losing control of plant reactivity caused by the automatic shifting of charging pump suction to the RWST as VCT level lowers.

Answer: A

Cognitive Level:

Analysis 3-PEO: Predict an Event or Outcome. The operator has to predict the system response based on a failure. This prediction is related to reason why they perform an action directed in the ARP

K/A:

022AK3.02 - Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Pump Makeup: Actions contained in SOPs and EOPs for RCPs, loss of makeup, loss of charging, and abnormal charging. RO/SRO IMP 3.5/3.8

Objective:

RO4-03-SED07.002 - In accordance with ARP-47043-L, "VCT Level High/Low," respond to a VCT Level High/Low Alarm

RO2-05-LP035.004 - DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Chemical and Volume Control System.

Reference:

ARP-47043-L, VCT Level High/Low, Step 4.
OPERXK-100-36, Analytical Part Flow – Chemical and Volume Control System
E-2023, Integrated Logic Diagram – Chemical & Volume Control Sys SH 1

Source:

Salem Exam – 2/99
Kewaunee Exam – 10/24/2000
ILT BANK RO2-05-LP035.004

Justification:

A CORRECT — VCT level will continue to lower, LD-27 is diverting and no automatic makeup. No automatic swap of charging pump suction to the RWST both LI-112 and LI-141 must both be less than 5% for automatic swap over to occur. VCT will empty the charging pumps will lose suction pressure and cavitate.

B INCORRECT — Auto makeup is driven from instrument LI-112 NOT LI-141. Since the instrument has failed high there will be no automatic makeup.

C INCORRECT — Letdown flow will not rise when LD-27 shifts to DIVERT. Letdown flow is diverted to the CVC holdup tanks. Letdown flow is controlled by the Letdown orifices and LD-10 and is not affected by the failure.

D INCORRECT — With the instrument failure high, charging pump suction will not automatically swap to the RWST.

7. Given the following:

    - A heat-up is in progress following a refueling shutdown.
    - Highest cold leg temperature indicates 205ºF.
    - RXCP ‘A’ is running.
    - Steam Generator ‘A’ & ‘B’ narrow range levels are greater than 20%.
    - RHR is being aligned for Split Train Mode per NOP-RHR-001, “Residual heat Removal System Operation”
    - SI-350B, CNTMT Sump B To RHR Pump B, failed to OPEN when directed to cycle the valve per NOP-RHR-001.

What additional failure would require action to place the plant in MODE 5 if NOT corrected within ONE hour?

A. SI-4A, RWST Supply to SI Pumps, failed CLOSED

B. SI-300A, RWST Supply to RHR pump A, failed CLOSED

C. RHR-1A, RCS Loop A Supply to RHR Pumps, failed CLOSED

D. ICS-2B, CNTMT Spray Pump B Suction From The RWST, failed CLOSED

Answer: B

Cognitive Level:

Analysis 3-SPK: Solve a Problem using Knowledge and its meaning. Must understand the SI, RHR, and ICS systems apply a failure to the systems to determine and solve what additional failure will result in the condition.

K/A:

025AA1.22 - Ability to operate and/or monitor the following as they apply to the Loss of Residual Heat Removal System: Obtaining of water from RWST for LPI system RO/SRO 2.9./2.8

Objective:

RO2-05-LP033.009 - Identify the Safety Injection Components with Technical Specifications.

RO2-05-LP033.010 - Explain the LCO operation, applicability and action requirements for the Technical Specifications associated with the Safety Injection System

References:

ITS Sections 3.4.6, 3.4.7, 3.5.3, 3.6.6
OPERM-217, Flow Diagram – Internal Containment Spray System
OPERXK-100-18, Flow Diagram – Residual Heat Removal System
OPERXK-100-29, Flow Diagram – Safety Injection System

Source:

New

Justification:

A INCORRECT Failure of SI-4A has no effect ECCS SI subsystem Safety Function because SI-4A and SI-4B are in parallel. Thus the failure of SI-4A closed does not result in any additional ECCS safety function loss.

B CORRECT Failure of SI-300A, causes the second RHR subsystem to be inoperable. Without a RHR subsystem operable one ECCS train is not operable. Condition 3.5.3 has action to restore in 1 hour. If cannot be restored in one hour, then be in in MODE 5 in 24 hours.

C INCORRECT Failure of RHR-1A removes one RHR loop from decay heat removal, there are still 2 RCS loops and RHR loop B for decay heat removal. (ITS 3.4.6)

D INCORRECT Failure of ICS-2B causes one train of containment spray to be inoperable. Condition entry has a 72 hour action to restore to operable status.

8. Given the following:

   - The unit is operating at 100% Rated Thermal Power.
   - A malfunction in the Component Cooling Water System has occurred.
   - PPCS Component Cooling Temperature T0621A is 105ºF and rising.

From the list below complete the following. If the component cooling water heat exchanger outlet temperature rises to _(1)_ , the correct operator actions would be to _(2)_.

(1)	(2)
A. 110ºF	manually start the standby component cooling water pump
B. 120ºF	trip the reactor and trip both Reactor Coolant pumps
C. 135ºF	isolate CC-300, Letdown Heat Exchanger Inlet Isolation Valve
D. 145ºF	isolate CC-610A/B, RXCP Therm Barr Comp Cooling Return Valve

Answer: B

Cognitive Level:

Memory 1-P: -Recall the setpoints for Reactor Coolant Pump Trip Criteria as a result of high Component Cooling water temperatures.

K/A:

026AA2.04 - Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The normal values and upper limits for the temperatures of the components cooled by CCW. RO/SRO IMP 2.5/2.9

Objective:

RO2-01-LP031.004 - DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Component Cooling Water System and the following major system components: RXCP Cooling Water.

Reference:

AOP-RC-005, Abnormal RXCP Operation, Step 1 and Foldout Page
ARP-47023-H, CC HX Outlet Temp High, Step 1

Source:

New

Justification:

Per ARP-47023-H, Component Cooling Heat Exchanger outlet temperature greater than or equal to 120ºF.

Step 1 is to Check PPCS Component Cooling Temperature - less than 120ºF. RNO states IF reactor is critical, then TRIP reactor and stabilize plant using EOPs, STOP RXCP and place in pullout.

Per AOP-RC-005, Abnormal RXCP Operation, Fold Out Page, if any of the following conditions listed below occur, then go to Step#1, PPCS Component Cooling Temperature T0621A - Greater Than or equal to 120ºF.

A INCORRECT — Parameter related to starting the standby component cooling water pump is related to a loss of pressure or flow. Starting a standby component cooling water pump is not directed for high CC temp.

B CORRECT — See above RXCP Trip criteria.

C INCORRECT — Actions to isolate unnecessary equipment due to high CC temperatures are contained in AOP-CC-0001. In this case, at power with letdown in service the letdown heat exchanger is not unnecessary. The value 135ºF is the annunicator setpoint for 47015-J, RXCP A/B total CC WTR OUTLET TEMP HIGH.

D INCORRECT — Annunciator 47014-J, Either RXCP Thermal Barrier return temp greater than or equal to 160ºF. If a thermal barrier is leaking actions are directed to isolate cooling return valves CC-610A/B.

9. Given the following:

   - The unit is at 100% Rated Thermal Power.
   - Pressurizer Pressure Control is in Automatic controlling pressure at 2235 psig.
   - PRZR Pressure Channel Control Selector switch is in the 2-1 position.
   - PT-429, Red Channel Pressurizer Pressure has Failed LOW.

With NO Operator action what prevents RCS pressure from exceeding the Technical Specification Pressure Safety Limit?

A. The Pressurizer Safeties will OPEN

B. ONLY PR-2A, Pressurizer PORV, will OPEN

C. ONLY PR-2B, Pressurizer PORV, will OPEN

D. BOTH PR-2A and PR-2B, Pressurizer PORVs will OPEN

Answer: A

Cognitive Level:

Comprehension 2-DR: Operator must assess two conditions and the effects that combination will have on the Pressurizer and plant.

K/A:

027G2.2.22 – Pressurizer Pressure Control System Malfunction: Knowledge of the limiting conditions for operations and safety limits RO/SRO IMP 4.0/4.7

Objective:

RO2-01-LP36B.002 - Describe the PRZR and PRT systems. Include the following in the description:

2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   b. heaters
   f. Power Operated Relief Valves

Reference:

E-2038, Integrated Logic Diagram – Reactor Coolant System
OPERXK-100-10, Flow Diagram – Reactor Coolant System

Source:

New

Justification:

A CORRECT The PRZR safeties will open because neither of the PRZR PORVs will make up the logic for opening so pressure will rise to the setpoint for the PRZR Safeties.

B INCORRECT PR-2A requires both the Control Channel for PRZR pressure, Red Channel (Channel 1 P-429) to be selected for control as indicated in the stem, and P-449 to be greater than the setpoint. Since P-429 failed low, PR-2A will not OPEN.

C INCORRECT PR-2B requires both the Protection Channel for PRZR pressure, White Channel (Channel 2 P-430) to be selected for protection as indicated in the stem, and P-429 to be greater than the setpoint. Since P-429 failed low, PR-2B will not OPEN.

D INCORRECT PR-2A requires both the Control Channel for PRZR pressure, Red Channel (Channel 1 P-429) to be selected for control as indicated in the stem, and P-449 to be greater than the setpoint. Since P-429 failed low, PR-2A will not OPEN.

PR-2B requires both the Protection Channel for PRZR pressure, White Channel (Channel 2 P-430) to be selected for protection as indicated in the stem, and P-429 to be greater than the setpoint. Since P-429 failed low, PR-2B will not OPEN.

10. Given the following:

- The unit was operating at 100% Rated Thermal Power when a Small Break LOCA occurred.
- Safety Injection initiated but the reactor trip breakers FAILED TO OPEN.
- The crew transitioned to FR-S.1, "Response to Nuclear Power Generation/ATWS".
- Subsequent actions by the Operators have been **UNSUCCESSFUL** in lowering Reactor NI power less than 20% before an Operator identifies that RCS subcooling is 15ºF and degrading rapidly.

What action is directed in FR-S.1 regarding the operation of the Reactor Coolant Pumps (RXCPs) AND what is the reason for the action?

Action	Reason
A. Maintain the RXCPs operating	Provide forced flow through the core for heat removal as the core voids.
B. Maintain the RXCPs operating	Reduce the effect of core voiding on the Source Range Nuclear Instrument indication.
C. Stop the RXCPs	Prevent excessive depletion of RCS inventory to maximize time the RWST is available.
D. Stop the RXCPs	Prevent damage to the pumps which allows for their use later in heat removal from the core.

Answer: A

Cognitive Level:

Comprehension 2-RI: The operator has to use the given conditions to determine the action and the effects of the action. Show the relationship of LOCA and ATWS with actions

K/A:

029EK1.01 - Knowledge of the operational implications of the following concepts as they apply to the ATWS: Reactor nucleonics and thermo-hydraulics behavior. RO /SRO IMP 2.9/3.1

Objective:

RO4-04-SED04.002 - Apply the following items as they relate to FR-S.1, Response to Nuclear Power Generation/ATWS

b. cautions

Reference:

BKG-FR-S.1, Background Response to Nuclear Power Generation/ATWS, Caution 1

FR-S.1, Response to Nuclear Power Generation/ATWS, First Caution and Step 20.

FR-C.1, Response to Inadequate Core Cooling, background, KPS step 12

BKG-E-1, Background Loss of Primary or Secondary Coolant, Foldout Page

Source:

New

Justification:

A CORRECT Background for the 1st caution in FR-S.1 states “RXCPs should not be tripped with reactor power greater than 5%.” The background gives the reason for heat removal from the core and states that the time requirements for RXCP tripping to remain within the small break LOCA design basis is not applicable.

B INCORRECT Core voiding does have an effect on the source range nuclear instrument reading, but this is not a concern at this time. Power Range Nuclear Instrument level less than 5% and Negative Intermediate SUR are used to check the reactor subcritical. FR-S.1 step 20.

C INCORRECT Actions applicable for FR-C.1 and large break LOCAs.

D INCORRECT Actions applicable if no ATWS. E-1 background for the actions are to prevent excessive core depletion which may lead to severe core uncovery.

11. Given the following:

- Reactor Power was originally at 100% Rated Thermal Power.
- Pressurizer level lowered rapidly and is currently 2% and lowering.
- A reactor trip and loss of Off-Site power occurred.

Which diagnosis indicates a Steam Generator (SG) tube rupture has occurred?

A. Reactor Thermal Power was rising before the reactor trip.

B. AFW flow to the ruptured SG is greater than AFW flow to intact SG.

C. SG pressure of the ruptured SG is rising in an uncontrolled manner.

D. Indicated steam flow was greater than feed flow before the reactor trip.

Answer: D

Cognitive Level:

Knowledge 1-P: Recall or recognize E-0 procedure diagnosis indications used to determine the transition to E-3, Steam Generator (SG) Tube Rupture

K/A:

038 G2.4.1 - Steam Generator Tube Rupture, Knowledge of EOP entry conditions and immediate action steps. RO/SRO IMP 4.6/4.8

Objective

RO4-04-LP002.008 - List the IMMEDIATE operator actions, given a reactor trip, is accordance with E-0.

RO4-04-LP028.001 - Discuss the following items as they relate to E-3 "Steam Generator Tube Rupture"

1. Purpose
2. Entry Conditions
3. Major Action Steps

Reference:

E-0, Reactor Trip or Safety Injection, Step 16

Source:

New

Justification:

A INCORRECT Reactor Thermal Power would go down as the heat transferred to the Steam Generators rises for the same RCS Tave. A faulted SG would cause power to rise.

B INCORRECT AFW flow is not used to diagnose SGTR. AFW flow to the ruptured SG would be lower than AFW to the intact S/G.

C INCORRECT SG pressure is not used to diagnose a SGTR.

D CORRECT Feed Flow would be less than steam flow before the trip. Step 16 of E-0 identifies if a SG is ruptured.

12. Given the following:

- The crew is performing FR-H.1, “Response to Loss of Secondary Heat Sink”
- The crew has OPENED both Pressurizer PORVs
- The crew has verified MCC-5262 energized by checking C-13, Condensate Bypass LP FW Heater, or Air Compressor 1A energized

Why is MCC-5262 verified energized?

A. Check a supply of compressed air is available for control of PR-2A and PR-2B, Pressurizer PORVs.

B. Check power is available to locally operate FW-2A and FW-2B, Feed Water Pump Discharge Isolation Valves.

C. Check that the Turbine Seal Oil system is operating to prevent the Hydrogen concentration in the Turbine from reaching explosive levels.

D. Check that a Waste Gas compressor is available to maintain Pressurizer Relief Tank Conditions to prevent rupturing of the Pressurizer Relief Tank Rupture Disk.

Answer: A

Cognitive Level:
Memory 1-B: Know the basis for verifying why MCC-5262 is verified energized.

K/A:
054AK3.04 - Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Actions contained in EOPs for loss of MFW. RO/SRO IMP 4.4/4.6.

Objective:
RO4-04-LP035.001 - Discuss the following items as they relate to FR-H.1, Response to Loss of Secondary Heat Sink.
c. Major Action Steps

Reference:
BKG-FR-H.1, Background Loss of Secondary Heat Sink, Step 7, 16, and 28
FR-H.1, Loss of Secondary Heat Sink

Source:
New

Justification:

A CORRECT — Reason is as stated in the background document for FR-H.1 step 28.

B INCORRECT — Power to operate FW-2A and FW-2B is not from MCC-5262. FW-12A/B are operated in Step 11 of the procedure for establishing main feedwater flow path and, in step 19.b during establishment of condensate flow path to a depressurized steam generator.

C INCORRECT — The Turbine Back Up Seal Oil pump is powered from MCC-5262, though hydrogen is a concern other procedures would be used to address the loss of seal oil and no indications were given for a loss of power to other bus.

D INCORRECT — Waste Gas Compressor is powered from MCC-5262 and the Waste gas system is connected to the PRT. Waste gas compressors will not help mitigate a loss of heat sink.

13. Given the following:

    - A loss of Off-Site power has occurred.
    - Bus 6 is LOCKED OUT.
    - Diesel Generator 'A' has tripped.
    - Annunciator 47091-B, Diesel Gen A Mech Lockout, is LIT
    - The local operator reports each of the alarms listed below are LIT and Diesel Generator ‘A’ Primary Tank Air Pressure is 190 psig.

Which of the following alarms, if it alone were cleared, would allow a restart of Diesel Generator 'A'?

A. Diesel Generator annunciator DR101-11, High Water Temp.

B. Diesel Generator annunciator DR101-21, Low Lube Oil Level.

C. Diesel Engine Control Panel D-1A L7, Jacket Water Pressure, lamp.

D. Diesel Engine Control Panel D-1A L10, Low Air Pressure System #2 (Primary), lamp.

Answer: C

Cognitive Level:

Memory 1-I: Interlocks, set points, or system (singular) response

K/A:

055A1.06 - Ability to operate and/or monitor the following as they apply to a Station Blackout: Restoration of power with one ED/G. IMP RO/SRO 4.5/4.5

Objective:

RO2-03-LP42A.004 - DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Emergency Diesel Generator and TSC Diesel Generator Systems and the following major system components:

1. DG control switch
2. DG voltage control switch
3. DG speed control switch
4. DG output breaker selector switch
5. DG 1A engine and generator LOCAL/REMOTE switches
6. DG start sequence
7. DG trip
8. DG output breaker
9. DG stop sequence
10. DG SW control valves SW-301A and SW-301B
11. TSC DG trip
12. TSC DG stop sequence
13. TSC DG voltage restoration logic

Reference:

ARP-47091-B, Diesel Gen A Mech Lockout
ARP-D-1A-8, Jacket Water Pressure
ARP-D-1A, Low Air Pressure System #2
ARP-DR101-23, Low Air Pressure
ARP-DR101-11, High Water Temperature
ARP-DR101-21, Low Lube Level

Source:

Bank: ILT Initial Bank RO2-03-LP42A.004 1
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Justification:

A INCORRECT The ENGINE PRE-FAILURE alarm has 6 inputs from alarm conditions displayed on Panel D-1A. However, none of these provide for D/G lockout.

B INCORRECT Low lube oil level will provide for an alarm but the D/G does NOT trip/lockout on lube oil level.

C CORRECT DG lockout occurs: (1) engine overspeed; (2) low jacket water pressure (w/running conditions); (3) low lube oil pressure (w/running conditions); (4) Unit Start Failure; (5) Loss of DC control power. The lamp D-1A L7 is the indicator for the low jacket water pressure trip and to reset the alarm (once pressure is above setpoint), the Failure Reset pushbutton on Panel D-1A must be depressed.

D INCORRECT 200 psig Air Receiver pressure indicates that pressure has fallen to the point where engine start MAY NOT be possible, so the D/G is declared inoperable if the pressure falls below this value. NO lockout condition is associated with this condition. The DG will not start if D-1A L7 is in.

14. Given the following:

- The unit was operating at 100% Rated Thermal Power when the following indications were observed:

    - Service Water (SW) Header Train 'A' indicated 78 psig and lowering
    - Service Water Pumps 1A1, 1A2, and 1B1 are running
    - Circulating Water Pump ‘A’ TRIPPED
    - Circulating Water Pump 'B' TRIPPED
    - SW-4A, SW Header A to Turbine Bldg Hdr, is OPEN
    - SW-4B, SW Header B to Turbine Bldg Hdr, is CLOSED

Which of the following could cause ALL the above indications?

A. SW Strainer differential pressure indicates 6 psid.

B. A SW leak has occurred in the turbine building.

C. A Reserve Auxiliary Transformer Lockout

D. Forebay water level indicates 46%.

Answer: B

Cognitive Level:

Comprehension 2-DR: Recognize the indications and determine the cause as a potential loss of Service water due to leak in the Turbine Bldg.

K/A:

062AA 2.02 - Loss of Nuclear Service Water: Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: the cause of possible SWS loss. RO/SRO IMP 2.9/3.6

Objective:

RO4-03-LPD11.005 - In accordance with OP-KW-AOP-SW-001, Abnormal Service Water System Operation, Summarize the subsequent operator actions necessary to respond to the following:

1. Low Forebay Water level
2. Low SW Header Pressure from SW isolation
3. Low SW Header Pressure from Insufficient operating pumps
4. Low SW Header Pressure from System Leakage.

Reference:

AOP-SW-001, Abnormal Service Water System
ARP-47051-N, CW Pumps Flood Level Trip
ARP-47051-Q, Turbine Bldg Service Water Isolation
ARP-47054-P, SW Strainer Diff Press High
ARP-47052-P, Turbine Bldg SW Header Abnormal
ARP-47051-M, CW Pumps Low Low Level Trip
E-1636, Integrated Logic Diagram – Diesel Generator
E-1638, Integrated Logic Diagram – Diesel Generator Electric
E-1631, Integrated Logic Diagram – Service Water System
E-1630, Integrated Logic Diagram – Service Water System
E-1614, Integrated Logic Diagram – Circulating Water System
E-1633, Integrated Logic Diagram – Service Water System

Source:

New

Justification:

A INCORRECT — The SW strainer differential pressure that results in automatic actions is 5 psid. Also the only automatic action is initiation of backwash. The alarm is actuated at 8 psid.

B CORRECT — The lowering SW header pressure indicates a problem maintaining header pressure with three pumps running. Both CW pumps tripped indicate either a loss of forebay water level or Turbine building flooding. The service water valves SW-4A/B are in there normal alignment. The only answer that would result in the given indications is a SW leak in the Turbine Building.

C INCORRECT — A loss of RAT would result in the DG sequencing loads on the bus. This would not cause the CW Pumps to Trip.

D INCORRECT — The forebay water level that results in a trip of both CW pumps is less than or equal to 42%.

15. Given the following:

- The unit is at 100% Rated Thermal Power.
- IA-101, Instrument Air to Containment Isolation valve, is inadvertently CLOSED and NOT RE-OPENED.
- A small air leak exists in the Containment air header.

If air pressure is NOT restored to containment, which of the following will cause a reactor trip?

(Assume NO operator action)

A. Letdown isolation will result in HIGH Pressurizer level.

B. Closure of Main Feed Regulating Valves will result in LOW SG level.

C. Opening of Pressurizer Power Operated Relief Valves will result in LOW Pressurizer pressure.

D. Containment Fan Coil Units Emergency Discharge Dampers opening will result in a Reactor Coolant Pump TRIP due to overheating.

Answer: A

Cognitive Level:

Comprehension 2-RI: Recognizing the interaction between systems, including consequences and implications.

K/A:

006AK3.0.3 - Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Knowing effects on plant operation of isolating certain equipment from instrument air. RO/SRO IMP 2.9/3.4

Objective:

RO2-02-LP001.002 - DESCRIBE the Station and Instrument Air System, include the following in the description;

1. System flow path
2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:

Reference:

OPERM-213-1, Operation Flow Diagram – Station and Instrument Air System, Rev. CC
OPERM-213-8, Operation Flow Diagram – Station and Instrument Air System, Rev. K
OPERM-213-6, Operation Flow Diagram – Station and Instrument Air System, Rev. Q
000-SD-01, 1, System Description Station Air and Instrument Air System, Rev. 2, Section 3.7.4

Source:

KNPP Initial Bank RO2-02-LP001.002 #3

Justification:

A CORRECT	When Instrument air is lost to containment, after the air receiver pressure bleeds down, the letdown isolation valves fail closed. This will result in charging filling the Pressurizer and eventually cause a reactor trip.
B INCORRECT	The main feedwater regulating valves are outside containment.
C INCORRECT	Pressurizer PORVs fail closed.
D INCORRECT	Containment FCU emergency discharge dampers (RVB-150A/B/C/D) will fail open when the accumulators bleed down however, the RXCPs do not have an automatic trip on temperature.

16. Given the following:

- The unit was operating at 45% Rated Thermal Power when the reactor automatically tripped.
- Safety Injection automatically initiated.
- The Crew diagnosed a LOCA outside of Containment and transitioned to ECA-1.2, “LOCA Outside Of Containment”.
- The LOCA was isolated when the crew CLOSED SI-302A, RHR Pump A Injection To Reactor Vessel.
- RWST level continued to lower until the crew CLOSED SI-300A, RWST Supply To RHR Pump A.
- Plant Parameters after SI-300A, RWST Supply to RHR Pump A, was closed were:

Parameter	Value	Trend
Pressurizer Pressure	2210 psig	Stable
Pressurizer Level	30%	Rising
Cold Leg Temperatures	450ºF	Rising
RWST Level	273,000 gal (Usable)	Stable
Total AFW Flow	240 gpm	Slowly Lowering
Steam Generator ‘A’ Pressure	410 psig	Rising
Steam Generator ‘B’ Pressure	400 psig	Rising
Steam Generator ‘A’ Narrow Range Level	11%	Rising
Steam Generator ‘B’ Narrow Range Level	15%	Rising

Which of the following is NOT able to meet its Technical Specification required surveillances at this time?

A. RWST

B. RCS Loop 'A'

C. ECCS SI Train ‘A’ subsystem

D. ECCS RHR Train 'B' subsystem

Answer: C

Cognitive Level:

Analysis 3-SPK: Analyze the plant conditions to determine which loss of function of emergency system effects TS.

K/A:

W/E04EK1.1 - Knowledge of the operational implications of the following concepts as they apply to the LOCA Outside Containment: Components, capacity, and function of emergency systems. RO/SRO IMP 3.5/3.9

Objective:

ROI-01-LPTS1.005 - Given plant conditions, IDENTIFY the system components that are required by the ITS Bases to be OPERABLE to meet an LCO

RO2-05-LP033.006 - Identify the Safety Injection components with Technical Specifications

RO2-05-LP033.006 - Explain the LCO operation, applicability and action requirements for Technical Specifications with the Safety Injection System

References:

ITS, SR 3.5.4.1, 3.5.4.2, 3.4.5.1, 3.4.5.2, and 3.5.2.2

Source:

New

Justification:

A INCORRECT — SR 3.5.4.1 the RWST must have the required Level of $\geq$ 272,500 gallons and SR 3.5.4.2 Boron Concentration of $\geq$ 2500 ppm and $\leq$ 2650 ppm.

B INCORRECT — SR 3.4.5.1 Verify One RCS Loop is in operation. SR 3.4.5.2 Verify Steam Generator secondary side water levels are $\geq$ 5% for both RCS loops. RCS Loop A has RXCP with power, S/G level $\geq$ 5%, and ability to dump steam.

C CORRECT — SR 3.5.2.2 Verify that each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed or otherwise secured in position is in the correct position. ECCS SI Train 'A' subsystem cannot take suction from the Containment Sump because it requires the flow path through ECCS RHR subsystem 'B'. SI-300A and SI-302A are not in their correct position.

D INCORRECT — ECCS RHR Subsystem Train 'B' is complete and meets all the requirements of TS. All valves for the RHR subsystem still meet the requirements of SR 3.5.2.2.

17. Given the following:

- The unit was at 100% Rated Thermal Power when a large break LOCA occurred.
- The ECCS system is functioning as designed.
- Reactor Coolant Pumps are in PULLOUT
- Current Plant Parameters:

Parameter	Value	Trend
Wide Range RCS Pressure	25 psig	Stable
Core Exit Thermocouples	220ºF	Stable
RCS Cold Leg Loop Temperatures	150ºF	Stable
Steam Generator Pressures	400 psig	Stable
Steam Generator 'A' Narrow Range Level	2%	Slowly Rising
Steam Generator 'B' Narrow Range Level	4%	Slowly Rising
Total AFW Flow to Both Steam Generators	220 gpm	Stable
Containment Pressure	25 psig	Slowly Lowering
Pressurizer Level	0 %	---------
RVLIS Train 'A' & 'B'	85%	Slowly Lowering
RCS Void Fraction Train 'A' & 'B'	-10%	Stable
RWST Level	50%	Slowly Lowering

What is the current mechanism by which core decay heat is being removed?

A. AFW flow.

B. RCS break flow.

C. forced two phase flow through the core.

D. heat transfer to Component Cooling from RHR.

Answer: B

Cognitive Level:

Analysis 3-SPK: Analyze the plant conditions to determine method of decay heat removal

K/A:

W/E05EK2.2 - Knowledge of the interrelations between the Loss of Secondary Heat Sink and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility. RO/SRO IMP 3.9/4.2

Objective:

RO4-05-LP005.003 - Describe the initial trends of major primary and secondary parameters following a DCRI event

References:

BKG-FR-H.1, Background Response to Loss of Secondary Heat Sink, Step 1
Steam Tables (1967)
ES-1.3 Transfer to Containment Sump Recirculation

Source:

Bank Master Bank 011EK3.12 1 - Modified

Justification:

A INCORRECT — During a large break LOCA the Steam Generators are Thermally disconnected from the Steam Generators.

B CORRECT — RCS break flow is the method for heat removal during Large Break LOCA.

C INCORRECT — The pressure and temperature of the CETs are such that subcooled conditions exist in the core.

D INCORRECT — With RWST level at 50% the RHR system is in injection mode from the RWST. Transfer to containment sump recirculation per ES-1.3 does not begin until RWST level of 37%.

18. Given the following:

- The unit is operating at 100% Rated Thermal Power.
- The crew entered AOP-EG-001, "Abnormal Grid Conditions" when 22 OCB Tripped OPEN and grid instability caused the 138KV voltage to lower to 142 KV.

Based on plant Operating Experience what controller operation may occur and what action is directed by AOP-EG-001 to mitigate the consequences of the controller operation?

	Controller Operation	Operation
A.	Turbine Control Valve Cycling	Disengage the Valve Position Limiter
B.	SW-1306A & SW-1306B,CC HX Temp Control Valves, fail OPEN	Position LD-14, LD Demin High Temp Divert Valve, to V.C.TK
C.	C-701, Condensate Recirculation Control Valve, fails CLOSED	Take manual control of C-701 in the Control Room
D.	Generator #1 Voltage Regulator maintains a constant output voltage causing MVARs to LOWER	Raise Generator #1 Base Adjuster

Answer: B

O/E:

Corrective Action Program Condition Report (CR) 327317, 22 OCB Tripped Open, reports an event at KPS where grid instability occurred with voltages lowering to ~ 142 KV. One of the conditions that occurred during this event was SW-1306A and SW-1306B failed open.

Cognitive Level:

Comprehension 2-RW: Re-Wording / re-phrasing information based on procedural direction to place LD-14, Demin High Temp Divert Valve, control switch to V.C.TK. This action bypasses the demins. The reason is based upon overcooling of CVCS letdown and the demineralizers increased affinity for boron in this condition.

K/A:

077AK2.04 - Knowledge of the interrelations between Generator Voltage and Electrical Grid Disturbance and the following: Controllers, positioners. RO/SRO Imp 3.0/3.0

Objective:

RO2-02-LP002.004 - DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Service Water System and the following major system components:
SW pumps; SW pump preferred selector switch; Traveling water screens; Strainers; SW isolation valves (SW-3A/B); TB header isolation valves (SW-4A/B); AB header isolation valves (SW-10A/B); SW to AFW pumps (SW-502, SW-601A/B); Containment fan coil SW return valves (SW-903A/B/C/D); SW to CCW HX control valve (SW-1306A/B); SW emergency supply to CCW (SW-1400); SW to EDG (SW-301A/B); SW to Main Generator(SW-2602).

RO2-03-LP043.005 - EXPLAIN the purpose of the following procedures used to govern the normal, abnormal, and emergency operation of the Electrical Generation System
OP-KW-NOP-TB-001, Turbine and Generator Operation
OP-KW-NOP-GE-001,Operation of the Generator Hydrogen, Seal Oil, and Exciter Cooling System
OP-KW-AOP-EG-001, Abnormal Grid Conditions
Generator Annunciator Response Procedures

Reference:

AOP-EG-001, Abnormal Grid Conditions
CAP CR327317 - 22 OCB Tripped Open
E-1615, Integrated Logic Diagram – Condensate System

Source:
New

Justification:

A INCORRECT	With cycling frequency the crew is directed to engage the VPL. This would prevent changes in grid frequency from affecting the Turbine Load.
B CORRECT	Step 2 of AOP-EG-001 directs if SW-1306A and SW-1306B failed open to place LD-14 in V.C.TK position. This action bypasses the letdown demins. With SW-1306A/B open excessive cooling of CC occurs. Since CC provides cooling to the CVCS letdown (Letdown HX), CVCS is also cooled. More boron is removed from the cooler coolant flow by the CVCS demins. The demins are bypassed to prevent/minimize this dilution of the RCS.
C INCORRECT	C-701 is a fail open valve.
D INCORRECT	To control MVARs the operators would use Generator #1 Voltage Adjuster not the base adjuster.

19. Given the following:

- The reactor spuriously tripped from 100% Rated Thermal Power.
- All of the control rods in Control Bank ‘B’ remained fully withdrawn after the reactor trip signal.
- CVC-440, Emergency Boration To Charging Pumps, will NOT OPEN from the Control Room.
- The Unit Supervisor directs the Operator to perform the emergency boration per the RNO for failure of CVC-440 to operate from the control room.

Which of the following component manipulations states the correct system response when establishing the emergency boration flow path?

	Component Manipulation	System Response
A.	OPEN CVC-403, Boric Acid To Blender	Directs boric acid to the Boric Acid Blender and isolates Reactor Makeup Water from the Boric Acid Blender by causing MU-1022, Reactor Makeup Water to Blender, to CLOSE on interlock
B.	CLOSE CVC-405, Boric Acid Emergency Suct Isol	Isolate boric acid flow to the Boric Acid Blender and establishes a boric acid flow path directly from the Boric Acid Storage Tanks to suction of the charging pumps
C.	CLOSE CVC-406, BA Blender to VCT	Isolates boric acid flow from the Boric Acid Blender to the VCT and causes CVC-405, Boric Acid to Emergency Suct Isol, to OPEN on interlock
D.	OPEN CVC-408, BA Blender To Charging Pumps	Stops a Boric Acid Flow Deviation from automatically CLOSING CVC-408, BA Blender To Charging Pumps

Answer: D

Cognitive Level:

Comprehension 2-DR: Recognize the relationship between valve controls, and system logic to determine which valve manipulation will allow an emergency boration

K/A:

024AA1.13 - Ability to operate and/or monitor the following as they apply to the Emergency Boration: Boric acid flow controller. RO/SRO IMP 3.2/3.0

Objective:

RO2-05-LP035.004 - Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with Chemical and Volume Control System.

References:

AOP-CVC-001, Emergency Boration
E-2024, Integrated Logic Diagram – Chemical and Volume Control Sys
OPERXK-100-36 Operation Flow Diagram – Chemical and Volume Control System

Source:

New

Justification:

A INCORRECT CVC-403 does direct boric flow to the boric acid blender, it does not cause MU-1022 to go closed on interlock. If performing a normal boration with MU-1022 in Auto, MU-1022 is closed.

B INCORRECT CVC-405 bypasses the Boric Acid Blender and directs boric acid from CVC-403 to the suction of the Charging pumps, it does not bypass CVC-403.

C INCORRECT CVC-406 does isolate flow to the VCT but does not cause CVC-408 to OPEN on interlock.

D CORRECT The procedure directs the operator position CVC-408 control switch to open which prevents the valve from closing on a flow deviation.

20. Per AOP-RC-004, "Steam Generator Tube Leak", which of the following is the LARGEST stable RCS to Steam Generator Leak rate which does NOT require isolation of the leaking Steam Generator?

(Stable is defined as a constant rate for greater than 60 minutes)

A stable leak rate of . . .

A. 10 gallons per day.

B. 25 gallons per day.

C. 80 gallons per day.

D. 150 gallons per day.

Answer: C

Cognitive Level:
Memory 1-B: Bases or Purpose. The operator has to demonstrate knowledge of the SG leak rate which requires steam generator isolation

K/A:
037AA2.11 - Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: When to isolate one or more S/Gs. RO/SRO IMP 3.8/3.8

Objective:
RO4-03-LPD07.006 - In accordance with AOP-RC-004, "Steam Generator Tube Leak," and AOP-GEN-002, "Rapid Power Reduction," Summarize the subsequent operator actions that necessary to respond to a primary to secondary leak in any steam generator

RO4-03-SED07.009 - In accordance with AOP-RC-004, "Steam Generator Tube Leak," and AOP-GEN-002, "Rapid Power Reduction," Respond to a primary to secondary leak in a Steam Generator.

Reference:
AOP-RC-004, Steam Generator Tube Leak

Source:
New

Justification:

A INCORRECT At 10 gallon per day the requirement is for increased monitoring and increase setpoints on associated radiation monitors.

B INCORRECT Leakage greater than or equal to 5 gallons per day and less than 30 gallons per day is entry into action level 1. Action level 1 requires, increased monitoring, increasing the radiation monitor setpoints, increase sampling of the secondary, and trending the leak rate with reports to plant management.

C CORRECT Leakage greater than or equal to 75 gallon per day but less than 100 gallons per day (stable for greater than 1 hour) requires entry into action level 2. Action level 2 requires a plant shutdown per GOP procedures without isolation of the leaking Steam Generator.

D INCORRECT Leakage greater than or equal to 100 gallons per day is entry into action level 3. Action level 3 requires the operator to shutdown and isolate the leaking steam generator. 150 gallons per day is the Technical Specification number for primary to secondary leakage per Steam Generator.

21. Given the following:

    - Power is at 7% by Nuclear Instruments during a plant startup.
    - Intermediate Range NIS channel N35 Instrument Power has FAILED.
    - The following action of AOP-MISC-001, "Response to Instrument Failure", for the failed channel was completed:
        - The Channel Selector switch on the Startup Rate Section of the Comparator and Rate Drawer was POSITIONED to N36.
    - The Level Trip switch on the N35 drawer was POSITIONED to BYPASS for corrective maintenance.

Which of the following list of Annunciators and Permissive Lights would be consistent with these plant conditions and switch manipulations?

1. 47033-L, NI System SR/IR Trip Bypassed
2. 47032-N, IR High Flux Rod Stop
3. Permissive Light 44905-0602, Auto Rod WDL Block P-2
4. Permissive Light 44905-0202, IR Blocked

A. 1 and 3 LIT

B. 1 and 4 LIT

C. 2 and 3 LIT

D. 2 and 4 LIT

Answer: A

Cognitive Level:

Comprehension 2-RI: Recognize the relationship and plant interactions between RPS, Rod Control, & NIs

K/A:

033G.2.4.46 - Loss of intermediate Range: Ability to verify that the alarms are consistent with plant conditions RO/SRO IMP 4.2/4.2

Objective:

RO2-05-LP048.004 - Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Excore Nuclear Instrument System and the following major system components

1. Excore NIs Source Range Detectors and Drawers
2. Excore NIs intermediate Range Detectors and Drawers
3. Excore NIs Power Range Detectors and Drawers
4. Excore Nis Control Board indicators Recorders
5. Comparator and Rate Drawer
6. Scaler Timer Drawer
7. Detector Current Comparator Comparator
8. Audio Count Rate Drawer

Reference:

AOP-MISC-001, Response to Instrument Failure
ARP-47033-L, NI System SR-IR Trip Bypassed
ARP-47032-N, IR High Flux Rod Stop
E-2051-2, Integrated Logic Diagram – Source and Intermediate Range Nuclear Instrumentation
XK-100-156, Logic Diagram – Turbine Trip Runbacks and Other Signals (W/ Requirements)
GOP-105, Startup from MODE 2 to 35% Power

Source:

New

Justification:

A CORRECT Both annunciator and permissive will be lit.

B INCORRECT Annunciator lit, permissive not lit.

C INCORRECT Annunciator not lit, permissive lit.

D INCORRECT Both annunciator and permissive not lit.

22. Which of the following is an operational implication associated with Area Radiation Monitors R-40 and R-41, Containment Hi Level Radiation Monitors?

R-40 and R-41 contain . . .

A. a GM tube with a high level of sensitivity to detect RCS leakage as required by Technical Specifications.

B. an Ion Chamber that is required post-accident and is used to determine Adverse Containment conditions.

C. a Scintillation detector with high sensitivity which has installed insulation to allow for operation during a Design Basis Accident.

D. a Solid State Alpha detector which can discriminate in areas with high background radiation and initiates a Containment Vent Isolation.

Answer: B

Cognitive Level:

Memory 1-F: Recall the limitations for Containment High Level Rad Monitors.

K/A:

061AK1.01 Knowledge of the operational implications of the following concepts as they apply to Area Radiation Monitoring (ARM) System Alarms: Detector limitations. RO/SRO IMP 2.5/2.9

Objective:

RO2-01-LP045.002 - Describe the Radiation Monitoring System to include the following in the description: Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components.

Reference:

E-0, Reactor Trip or Safety Injection, Foldout page actions
USAR - 11.2-38 and 39
ITS, Section 3.3.3, Function 10, Containment Area High Range (Reg Guide 1.197)
E-2021 - Integrated Logic Diagram – Radiation Monitoring
SDBD-KPS-RM Radiation Monitoring System
ITS, Section 3.4.15, RCS Leakage Detection Instrumentation

Source:

New

Justification:

R-40 and R-41 contain an Ion Chamber with a range of 10E-0 to 10E-8 R/hr. These instruments are designed and qualified to function post-LOCA and operate in the containment post accident environment.

A INCORRECT See above, the range of the detector prevents detection of minor leakage. The RCS leakage detection Rad Monitors per Tech Specs are R-11 and R-12.

B CORRECT See above.

C INCORRECT R-40 and R-41 are not scintillation detectors with installed insulation. They are required to operate after a Design Basis Accident.

D INCORRECT R-40 and R-41 are not Solid State Alpha detectors and do not initiate a Containment Vent Isolation.

23. Given the following:

- The unit is at 100% Rated Thermal Power.
- The Control Room is required to be evacuated due to a fire in the Alternate Fire Zone.
- The crew is performing the actions of AOP-FP-002, "Fire in Alternate Fire Zone."
- The reactor failed to trip from the Control Room.

What actions are required to ENSURE the reactor is tripped per AOP-FP-002?

A. Locally place Rod Drive MG Set Motor & Generator Circuit Breaker Control Switches to TRIP.

B. Locally TRIP the reactor at the DSP using the DSP Manual Reactor Trip Pushbutton.

C. Locally OPEN the reactor trip and bypass breakers.

D. Locally OPEN bus 33 and 43 supply breakers.

Answer: C

Cognitive Level:
Memory 1-P: Knowledge of Procedure Steps
K/A:
068AK2.02 - Knowledge of the interrelations between the Control Room Evacuation and the following: Reactor trip system. RO/SRO IMP 3.7/3.9
Objective:
RO4-03-LPD21.004 - Discuss the required actions prior to evacuating the control room
Reference:
AOP-FP-002, Fire in Alternate Fire Zone
FR-S.1, Response to Nuclear Power Generation/ATWS
E-0, Reactor Trip or Safety Injection
Source:
New

Justification:

A INCORRECT The actions for the Rod Drive MG set motor and generator circuit breaker control switches are listed in FR-S.1. The crew does enter E-0 per a note in AOP-FP-002.

B INCORRECT There is no manual reactor trip pushbutton at the DSP.

C CORRECT Per the RNO step of AOP-FP-002, if the reactor does not trip then the operator is directed to locally open the reactor trip and bypass breakers.

D INCORRECT There are no actions to locally open Bus 33 and 43 breakers, the actions listed in AOP-FP-002 directs the operator to open bus 33 and 43 breakers from the control room.

24. Given the following:

- FR-C.1, "Response To Inadequate Core Cooling" is in progress.
- Both RXCPs were stopped per FR-C.1 in preparation for SG depressurization to atmospheric pressure.
- Actions to reduce Core Exit Thermocouple temperatures have been unsuccessful.
- FR-C.1 directs the subsequent re-start of the RXCPs in available loops.

What is the purpose of the RXCP re-start?

A. To ensure the SI Accumulators are fully discharged and the coolant is circulated.

B. To raise RCS pressure to collapse voids which will allow Safety Injection Pump flow.

C. To force two phase or single phase flow through the core to provide temporary cooling.

D. To use RXCP discharge pressure to provide PRZR spray flow to reduce RCS pressure.

Answer: C

Cognitive Level:
Memory 1-B: Recall the purpose of RXCP restart during FR-C.1.

K/A:
074EK3.07: Knowledge of the reasons for the following responses as they apply to the Inadequate Core Cooling: Starting up emergency feedwater and RCPs. RO/SRO IMP 4.0/4.4

Objective:
RO4-04-LP024.002: SUMMARIZE purposes or bases of the following items as they relate to FR-C.1, "Response to Inadequate Core Cooling."

- Notes
- Cautions
- Attachments
- Procedure Transitions
- All Procedure Steps

Source:
New

Reference:
FR-C.1, Inadequate Core Cooling
BKG-FR-C.1, Background Inadequate Core Cooling

Justification:

Per background for FR-C.1, KPS step 17: "Check if RXCPs should be started" The actions of this step may provide temporary core cooling until some form of makeup flow to the RCS is established or one of the above items is restored. To temporarily restore core cooling, the operator is instructed to start RXCP's one at a time... The RXCPs should force two phase flow through the core, temporarily keeping it cool. Even single phase forced steam flow will cool the core for some time provided RXCPS can be kept running and a heat sink is available.

A INCORRECT The SI accumulators are fully discharged prior to or during the initial depressurization in FR-C.1.

B INCORRECT The RXCPs are not used to collapse voids. They may sweep voided areas clear to ensure a flow path to a S/G. The intent is for cooling. There has been a failure of Safety Injection Pumps to get to this point in the procedure.

C CORRECT See above excerpt from FR-C.1 background.

D INCORRECT RXCP discharge pressure would allow for Pressurizer spray flow, however this is not the purpose of starting the RXCP.

25. Given the following:

- The unit is at 100% Rated Thermal power.
- The following annunciators are LIT:
  - TLA-15, RMS Above Normal
  - 47012-B, High Radiation Indication Alert
- Radiation Monitors indicated rising radiation levels in the following sequence:
  - 1st R-9, RCS Letdown Monitor
  - 2nd R-4, Charging Pump Room Monitor
  - 3rd R-13 and R-14, Aux Bldg Vent Exhaust Monitors
- All other Radiation Monitors indicate normal radiation levels

What event does the above sequence of radiation monitor alarms indicate?

A. LOCA outside containment

B. High Reactor Coolant Activity

C. RCS Leak in CVCS letdown line

D. Charging Pump Relief Valve Lifting

Answer: B

Cognitive Level:

Analysis 3-SPK: Solve a problem using Knowledge and its meaning. The operator must interpret the symptoms and discern a possible cause based on the order of the symptoms

K/A:

076AA1.04 - Ability to operate and/or monitor the following as they apply to High Reactor Coolant Activity: Failed fuel-monitoring equipment. RO/SRO IMP 3.2/3.4

Objective:

RO4-03-LPD08.001: DESCRIBE the purpose, symptoms, immediate operator actions, and automatic actions associated with the following procedures: OP-KW-AOP-RC-003, High Reactor Coolant Activity.

Reference:

AOP-RM-001, Abnormal Radiation Monitoring System, step 10.
AOP-RC-003, High Reactor Coolant Activity, entry conditions

Source:

New

Justification:

A INCORRECT A LOCA outside containment would cause R-13/R-14 to alarm, however R-9 would not alarm. Plausible distracter due to leak into the aux bldg causing aux bldg exhaust alarms.

B CORRECT High Reactor Coolant activity would first be seen on R-9, the RCS letdown Monitor. This would then cause an alarm in the Charging Pump room due to Charging Pump leakoff which would then result in R-13 and R-14, Aux Bldg Exhaust Monitors radiation indication increase.

C INCORRECT A leak in the CVCS line would not cause R-9 to indicate rising radiation levels. This is a plausible distracter because all of the other stated Radiation Monitors would indicate rising radiation levels.

D INCORRECT A CCP relief valve lifting would not cause R-4 or R-9 to indicate rising radiation levels. This is a plausible distracter because the student has to know where the charging pump relief discharges to.

26. Given the following:

- Small Break LOCA has occurred.
- The crew is performing ES-1.2, "Post LOCA Cooldown and Depressurization".
- RCS pressure is 350 psig and stable.
- Both RHR pumps were stopped and placed in AUTO as directed in ES-1.2.
- The LOCA is still in progress.

After the Safety Injection signal was RESET, the RHR pumps were STOPPED to . . .

A. conserve RWST inventory.

B. reduce the rate of RCS cooldown.

C. prevent damage to the RHR pumps.

D. minimize component cooling heat load.

Answer: C

Cognitive Level:

Comprehension 2-DR: Must determine why the RHR pumps were stopped. The student has to realize that the 350 psig RCS pressure stated in the question is greater than the shutoff head of the RHR pumps. The pumps will only have recirculation flow, thus could be damage from overheating.

K/A:

W/E03A2.2 - Ability to determine and interpret the following as they apply to the LOCA Cooldown and Depressurization: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments RO/SRO IMP 3.4/4.1

Objective:

RO4-04LP019.002 - Summarize purposes or bases of the following items as they relate to ES-1.2 "Post LOCA Cooldown and Depressurization"
e. All procedure Steps

References:

ES-1.2 Post LOCA Cooldown and Depressurization
BKG-ES-1.2 Background Post LOCA Cooldown and Depressurization

Source:

New

Justification:

A INCORRECT At RCS pressure of 350 psig the RHR pumps will not be injecting RWST water into the RCS. 350 psig is greater than the shutoff head of the RHR pumps.

B INCORRECT The rate of cooldown is not the reason for stopping the RHR pumps. The procedure does direct the operator to cooldown the plant at a rate not to exceed a 100ºF/hr but the cooldown rate is controlled by the rate of steam release not the rate of injection flow.

C CORRECT The RHR pumps should be stopped to prevent damage since only have minimal flow through the RHR pumps.

.

D INCORRECT Component Cooling heat will be reduced but is not the reason for securing the RHR pumps.

27. Given the following:

- During implementation of FR-P.1, "Response To Imminent Pressurized Thermal Shock Condition", SI-20B, Accumulator 'B' Isolation Valve, failed to CLOSE.

- The Unit Supervisor directs NG-108B, Nitrogen Supply to Accumulator 'B', OPENED AND then NG-110, ACMTR VENT Control, OPENED for venting of Accumulator 'B'.

How does an operator OPEN NG-108B AND NG-110?

	NG-108B	NG-110
A.	Locally in the Gas Bottle Storage area	Locally in the Gas Bottle Storage area
B.	Placing its control switch on Mechanical Vertical 'A' to OPEN	Locally in the blowdown tank area
C.	Adjusting its controller on Mechanical Control Console 'C' to 100%	Placing its control switch on Mechanical Vertical 'A' to OPEN
D.	Placing its control switch on Mechanical Control Console 'C' to OPEN	Adjusting its controller on Mechanical Control Console 'C' to 100%

Answer: D

Cognitive Level:
Memory 1-S: Structures and location. The student has to demonstrate knowledge of the valve, location and operation.

K/A:
W/E08 G 2.1.30 - Overcooling (PTS) Ability to locate and operate components, including local controls. RO/SRO IMP 4.4/4.0

Objective:
RO2-05-LP033.004 - Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Safety Injection System and the following major system components:

3. Safety Injection Accumulators

References:
OPERXK-100-28, Operation Flow Diagram – Safety Injection System
FR-P.1 Response to Imminent Pressurized Thermal Shock
N-SI-33-CL, Safety Injection System Pre-Startup Checklist

Source:
New

Justification:

A INCORRECT Both of the valves are located in containment. Plausible a number of valves are located in this area including NG-100A Nitrogen Supply to the accumulators.

B INCORRECT NG-108B is operated from Mechanical Control Console 'C'. On Mechanical Vertical 'A' there a grouping of valves used for sampling (Group 2 containment isolation), and vents and drains. NG-110 is operated from the containment. Plausible because NG-300 Nitrogen to the PRT, NG-303 Nitrogen to PRT, NG-310 Nitrogen to PRT Vent, NG-190 Nitrogen to Caustic Storage Tank Level are located in the blowdown area.

C INCORRECT NG-108B is operated by a control switch not a controller. NG-110 is operated by a controller and is located on Mechanical Control Console 'C'.

D CORRECT NG-108B is located on Mechanical Control Console 'C' and operated by a control switch. NG-110 is located on Mechanical Control Console 'C' and is operated by a controller.

28. What is a function of the Reactor Coolant Pump Flywheel?

A. Ensures sufficient flow through the core during normal operations to maintain the DNB margin.

B. Prevent overheating of RXCPs seals in an idle loop due to reverse rotation of the Reactor Coolant Pump.

C. Maximize loop flow during natural circulation by ensuring sufficient starting inertia to prevent reverse rotation.

D. Provide inertia for a slow coast down or reduction of reactor coolant flow so that DNB margin is maintained.

Answer: D

Cognitive Level:

Knowledge 1-B: The operator must recall the purpose/function of the flywheel.

K/A:

003K5.01 – Knowledge of the operational implications of the following concepts as they apply to the RCPS: The relationship between the RCPS flow rate and the nuclear reactor core operating parameters (quadrant power tilt, imbalance, DNB rate, local power density, difference in loop T-hot pressure). RO/SRO IMP 3.3/3.9

Objective:

RO2-01-LP36A.002 - Describe the Reactor Coolant Pumps. Include the following in the description

- Function/Purpose, design basis, operating characteristics, and physical location as appropriate for the flowing major components
  a. Reactor Coolant Pump Motor
     1. Flywheel

Reference:

SDBD-KPS-RC, System Design Basis Document for Reactor Coolant System
000-SD-36, System Description Reactor Coolant System

Source:

Modified, LXR Master bank, 003 2.1.28 001

Justification:

A INCORRECT The pump is what maintains the flow rate through the core, not the flywheel during normal operations.

B INCORRECT The flywheel does not prevent reverse rotation and if the pump did rotate backwards this would not cause heating of the seals.

C INCORRECT The anti-reverse rotation device is what stops the pump from rotating.

D CORRECT This is the Correct Answer.

29. Given the following:

- The unit has been operating at 75% Rated Thermal Power for the past 113 hours
- Generator output is 450 MW electric
- Rod Control is in AUTO
- Charging Pump 'A' is in AUTO at 25% demand
- Charging Pump 'B' is in MANUAL at 25% demand
- CC-302, Letdown Heat Exchanger Temperature Control Valve, is in AUTO and set at 120ºF
- Tavg is at program value for plant conditions
- Temperature Element TE-401B, Reactor Coolant Loop B-2 Cold Leg Temperature, FAILS HIGH (Red Channel)

With NO operator action, what is a control AND reactor protection function that are affected by the failure of TE-401B?

	Control Function	Reactor Protection Function
A.	Turbine Runback	Bistable for RCS Loop 'B' Tavg Low Low will be TRIPPED
B.	Charging flow will RISE	Loop 'B' Channel 1 Over Temperature Delta Temperature setpoint will LOWER
C.	Control Bank 'D' control rods will step OUTWARD	Loop 'B' Channel 1 Over Pressure Delta Temperature setpoint will LOWER
D.	CC-302, Letdown Heat Exchanger Temperature Control Valve, will OPEN	Bistable for Red Channel HI Tavg will be TRIPPED

Answer: B

Cognitive Level:

Comprehension 2-DR: Recognize the relationship between Cold leg temperature, controls and protection associated with other systems.

K/A:

004K5.16 - Knowledge of the operational implications of the following concepts as they apply to the CVCS: Source of Tave. and Tref. signals to control and RPS

Objective:

RO2-05-LP035.004 - Describe the operation, automatic actions, interlocks instrumentation, controls and alarms associated with the Chemical and Volume Control System

References:

E-2039, Integrated Logic Diagram – Reactor Coolant System
E-2044, Integrated Logic Diagram – Reactor Control and Protection System
E-2042, Integrated Logic Diagram – Reactor Control and Protection System
AOP-MISC-001, Response to Instrument Failure

Source:

New

Justification:

A INCORRECT A Turbine runback for OTDT may occur depending on plant conditions, but the bistable for LOW LOW Tavg will not be tripped, the failure will cause Tavg to rise not lower.

B CORRECT Charging flow will rise as Auctioneered High Tavg rises causing a rise in Pressurizer level to match Tavg and OTDT setpoint will lower.

C INCORRECT Control Rods would step inward for the failure. The OPDT setpoint would lower.

D INCORRECT CC-302 position would close as charging flow rises and temperature from the regenerative heat exchanger lowers (letdown flow remains the same) to maintain letdown temperature. Red Channel HI Tavg bistable would be tripped.

30. How does the Main Steam Reheat and Heater Drain System design prevent an uncontrolled RCS cooldown which could result in an automatic Safety Injection when the reactor trips?

A. The MSR reheat inlet valves are downstream of the Turbine Stop Valves, therefore, when the turbine trips steam is isolated to the MSRs.

B. A turbine trip causes an isolation of steam to the inlet of the MSR by isolating instrument air to MS-201A1/A2/B1/B2, Reheater Steam Control Valves.

C. A reactor trip signal causes isolation of MSR Drain valves by de-energizing the solenoid valve which isolates instrument air to the MSR Drain valves, these valves then fail CLOSED.

D. Differential pressure between the Reheater Drain Tank and Feedwater Heater 15A/B greater than 10 psid CLOSES HD-11A1/A2/B1/B2, Reheater Drain Tank to Feedwater Heater 15A/B isolating steam flow from the MSR.

Answer: B

OE:

On March 28, 2010 H.B. Robinson Unit 2 experienced a plant trip and automatic initiation of Safety Injection.

- An electrical fault caused the reactor to trip on low RCS loop flow.
- The electrical fault resulted in loss of power to several MCCs.
- As a result of this power loss, all Moisture Separator Reheater (MSR) Drain Tank Alternate Drain valves and MSR timer valves failed OPEN.
- This steam flow resulted in a cooldown of the RCS.
- Additionally power was unavailable to the MSR Shutoff valves preventing the valves from being closed from the Control Room.
- Operators started two charging pumps as directed by the emergency operating procedures, but failed to monitor Pressurizer Level or RCS Temperature.
- An automatic Safety Injection occurred due to low RCS pressure caused by the cooldown.

Cognitive Level:

Memory 1-F: Knowledge of how the MSR and HD system works.

K/A:

039K3.05 - Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: RCS. RO/SRO IMP 3.6/3.7

Objective:

RO2-02-LP011.004 – Describe the operation, automatic actions, interlocks, instrumentation, controls, and alarms associated with the Heater Drains and Bleed Steam System and the following Major components:

3. Moisture Separator Reheaters

References:

E-2057, Integrated Logic Diagram – Turbine System Sh 2
XK-101-24-1, Electro – Hydraulic Control & Lub System Diagram
E-1612, Integrated Logic Diagram – Heater & Moist Sep Drains, Bleed Steam
E-1629, Integrated Logic Diagram – Main Steam and Steam Dump System

Source:

New

Justification:

A	INCORRECT	MS-201A1/A2/B1/B2 are upstream of the Turbine Stop Valves.
B	CORRECT	MS-201A1/A2/B1/B2 will go closed on a turbine trip when fluid operated air pilot valve isolates instrument air the valves and they are closed by spring action.
C	INCORRECT	Isolation of the MSR drain valves will not prevent a reactor cooldown. Steam paths still exist via relief valves.
D	INCORRECT	Does not isolate steam from the MSR. A path still exists to the condenser if MSR reheater drain tank level rises.

31. Given the following:

    - A plant cooldown to MODE 5 is in progress.
    - The RCS is at 335ºF and 350 psig.
    - Pressurizer bubble exists, with heaters and spray used for pressure control.
    - RHR is aligned for split train operations per NOP-RHR-001, “Residual Heat Removal Operation.”
    - RHR-8A, RHR Heat Exchanger A Flow Control Valve is at 60% open.
    - An air leak develops on RHR-8A, RHR Heat Exchanger 'A' Flow Control Valve, completely depressurizing the actuator.
    - RHR, RCS, Charging and Letdown flows are stable prior to the failure.
    - RHR/CVC Spectacle Flange is OPEN.

With NO operator action, what is an effect of the complete depressurization of RHR-8A, valve actuator?

A. Regen Hx Letdown Temp (TI-127) LOWERS due to total flow to the Regenerative Heat Exchanger LOWERING.

B. RCS cooldown rate LOWERS when RHR-8A, RHR Heat Exchanger 'A' Flow Control Valve, CLOSES.

C. Charging Pump in AUTO speed RISES as level in the Pressurizer LOWERS due to total Letdown flow RISING.

D. Letdown Hx Outlet Flow (FI-134) LOWERS due to the LOWERING of the differential pressure between RHR and Letdown.

Answer: D

Cognitive Level:

Comprehensive 3-PEO: Has to understand how the failure affects RHR-8A valve operation and use that understanding to analyze system response based on determining RHR-8A position:

K/A:

005K6.03 - Knowledge of the effect of a loss or malfunction of the following will have on the RHRS: RHR heat exchanger. RO/SRO IMP 2.5/2.6

Objective:

RO2-05-LP034.001 - Describe the function/purpose, design basis, and operating characteristics of the Residual Heat Removal System

Reference:

NOP-RHR-001, Residual Heat Removal System
GOP-203, Shutdown From Hot Shutdown to RHR
OPERXK-100-35, Flow Diagram – CVCS
OPERXK-100-36, Flow Diagram – CVCS
OPERXK-100-18, Flow Diagram – Residual Heat Removal System

Source:

New

Justification:

A INCORRECT Flow from the RHR system enters the Letdown system downstream of the Regenerative Heat Exchanger. Flow from the RHR system does not flow through the Regenerative heat Exchanger. However with RHR-8A failing open, this will cause Letdown flow from RHR to lower (lower RHR-1A pump discharge pressure) and that will cause more flow from the RCS via the normal Letdown lineup, therefore T-127 should go up.

B INCORRECT RHR-8A fails open. The failing open of RHR-8A will increase the cooldown rate of the RCS.

C INCORRECT Total Letdown flow will lower. This would cause level in the Pressurizer to rise and not lower. Thus charging pump in Auto speed would lower not raise.

D CORRECT RHR-8A fails open with the loss of air to its actuator. This causes the discharge pressure of RHR pump A to lower. This causes the differential pressure between the Letdown and the RCS to lower. This lower differential pressure causes total Letdown flow to lower.

32. Given the following:

- The plant is in MODE 1.
- A Bus 5 Lockout occurs.
- The plant is immediately manually tripped due to unstable conditions.
- During the performance of the Immediate Actions of E-0, “Reactor Trip or Safety Injection”, a spurious Train ‘A’ Safety Injection signal was received.

Which of the following is the correct display of the Containment Isolation Active Panel in the Control Room following the spurious Train ‘A’ Safety Injection signal?

(Pictures of the Containment Isolation Active Panel on the next pages)

A. See Attached Page for selection A

B. See Attached Page for selection B

C. See Attached Page for selection C

D. See Attached Page for selection D

Answer: A

Cognitive Level:

Comprehension 2-RI: The operator determines the interaction between Safety Injection Actuation Signal, and AC, DC power supplies and air operated versus motor operated for ESFAS operated valves.

K/A:

006K2.04 – Knowledge of bus power supplies to the following: ESFAS-operated valves. RO/SRO IMP 3.6/3.8

Objective:

RO2-04-LP056.004 – DESCRIBE the operation, automatic actions, interlocks, instrumentation, control and alarms associated with the Containment System and the following major system components:

- Containment Isolation System (CI)(
- Containment Ventilation Isolation (CVI)

Reference:

E-1629, Integrated Logic Diagram – Main Steam and Steam Sump System
E-2010, Integrated Logic Diagram – Secondary Sampling System
E-2025, Integrated Logic Diagram – Chemical and Volume Control System
E-2040, Integrated Logic Diagram – Reactor Coolant System
E-2045, Integrated Logic Diagram – Component Cooling System
E-2075, Integrated Logic Diagram - Primary Sampling
NCL-CVC-001, Charging and Volume Control Prestartup Checklist
NCL-BT-001, Steam Generator Blowdown Treatment System Prestartup CL
N-CC-31-CL, Component Cooling System Prestartup Checklist
E-2191, Schematic Diagram – Solenoid Valves SV33030, 33092, 33274 (RC-422)
E-1924, Schematic Diagram – Solenoid Valves – SV-33017, 33158, 33159 (BT31A/32A)
E-233, Circuit Diagram DC Aux. and Emergency AC

Source:

New

Justification:

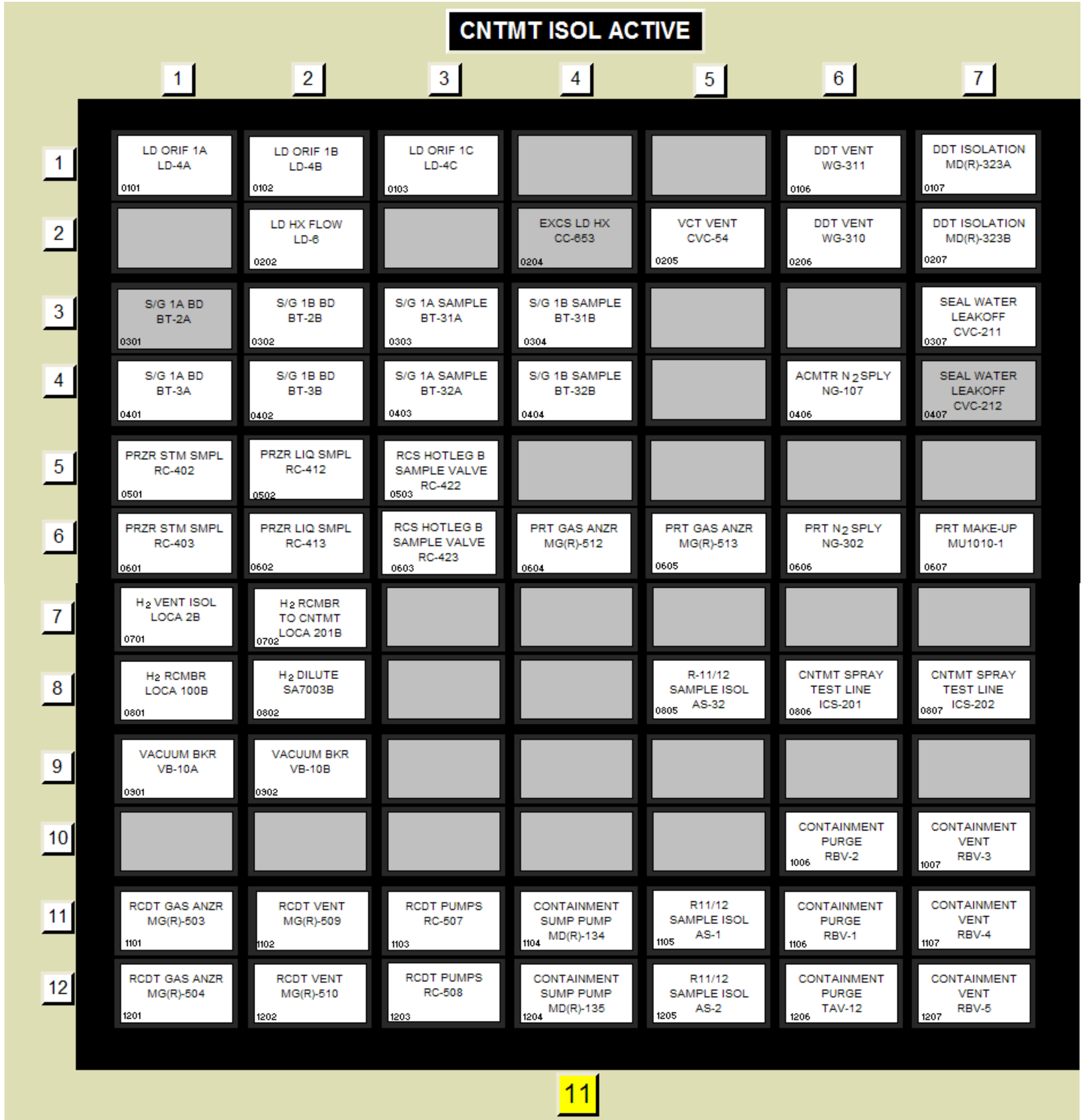
A CORRECT — BT-2A, CC-653, and CVC-212 are motor operated valves powered from Bus 5, therefore will not operate on a loss of power.

B INCORRECT — BT-2A, CC-653 are motor operated valves powered from Bus 5, therefore will not operate on a loss of power (light off), BUT MU-1010-1 is an air operated valve which should fail closed and has not lost power.

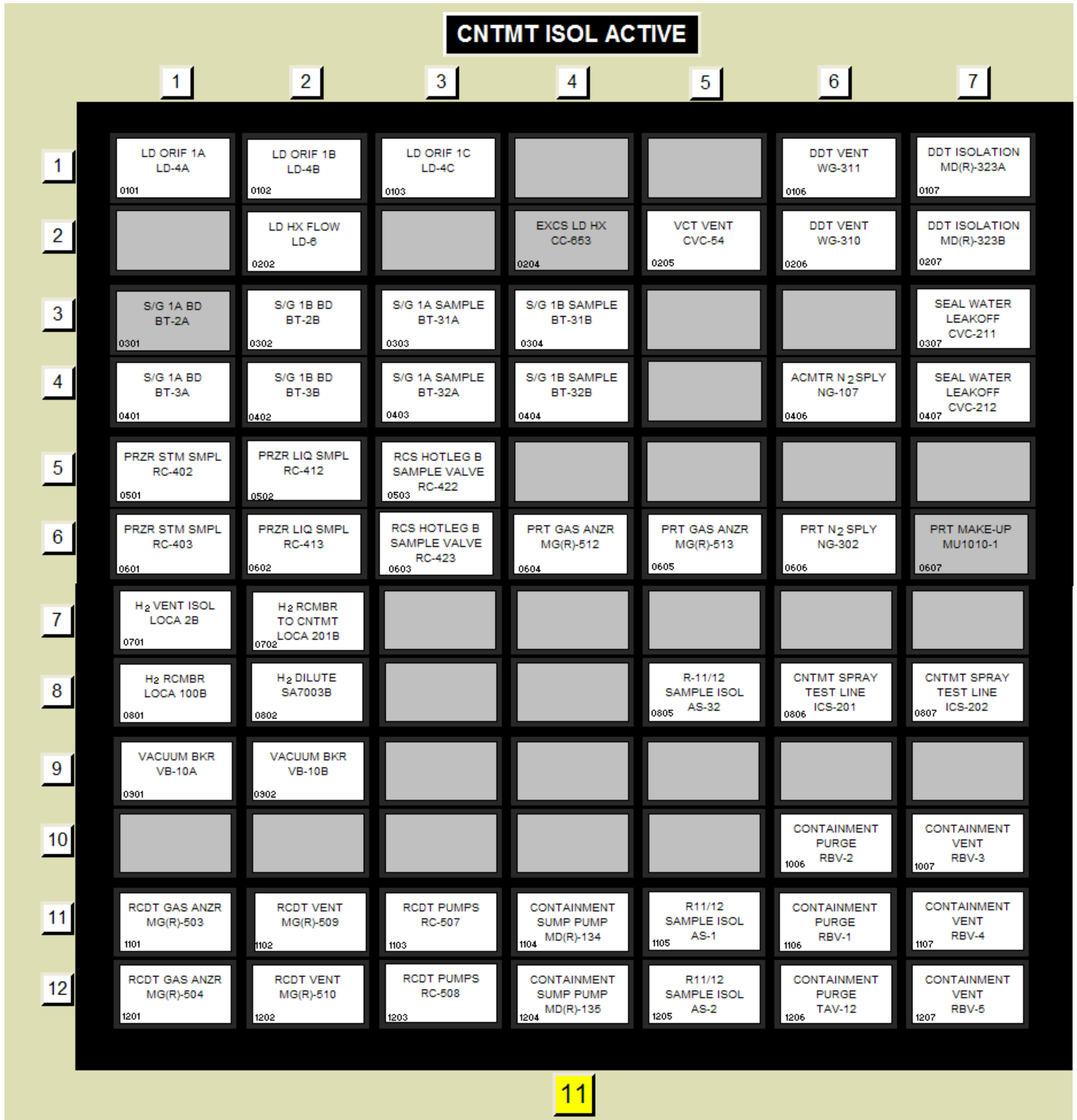
C INCORRECT — CVC-212, CC-653 are motor operated valves powered from Bus 5, therefore will not operate on a loss of power (light off), BUT BT-31A and BT-32A are solenoid operated valves powered from BRA-105 and should be closed due to failing closed on a loss of power (light on) and BT-2A is motor operated valve powered from Bus 5 and should be open (light off).

D INCORRECT — CVC-212, BT-2A are motor operated valves powered from Bus 5, therefore will not operate on a loss of power (light off), BUT RC-422 is a solenoid operated valve powered from BRA-104 and still has power on a loss of Bus 5, so should be closed (light on) and BT-2A is a motor operated valve powered from Bus 5 and should be open (light off).

A


CNTMT ISOL ACTIVE
1
2
3
4
5
6
7
1
LD ORIF 1A LD-4A 0101
LD ORIF 1B LD-4B 0102
LD ORIF 1C LD-4C 0103
DDT VENT WG-311 0106
DDT ISOLATION MD(R)-323A 0107
2
LD HX FLOW LD-6 0202
EXCS LD HX CC-653 0204
VCT VENT CVC-54 0205
DDT VENT WG-310 0206
DDT ISOLATION MD(R)-323B 0207
3
S/G 1A BD BT-2A 0301
S/G 1B BD BT-2B 0302
S/G 1A SAMPLE BT-31A 0303
S/G 1B SAMPLE BT-31B 0304
SEAL WATER LEAKOFF CVC-211 0307
4
S/G 1A BD BT-3A 0401
S/G 1B BD BT-3B 0402
S/G 1A SAMPLE BT-32A 0403
S/G 1B SAMPLE BT-32B 0404
ACMTR N2 SPLY NG-107 0406
SEAL WATER LEAKOFF CVC-212 0407
5
PRZR STM SMPL RC-402 0501
PRZR LIQ SMPL RC-412 0502
RCS HOTLEG B SAMPLE VALVE RC-422 0503
6
PRZR STM SMPL RC-403 0601
PRZR LIQ SMPL RC-413 0602
RCS HOTLEG B SAMPLE VALVE RC-423 0603
PRT GAS ANZR MG(R)-512 0604
PRT GAS ANZR MG(R)-513 0605
PRT N2 SPLY NG-302 0606
PRT MAKE-UP MU1010-1 0607
7
H2 VENT ISOL LOCA 2B 0701
H2 RCMBR TO CNTMT LOCA 201B 0702
8
H2 RCMBR LOCA 100B 0801
H2 DILUTE SA7003B 0802
R-11/12 SAMPLE ISOL AS-32 0805
CNTMT SPRAY TEST LINE ICS-201 0806
CNTMT SPRAY TEST LINE ICS-202 0807
9
VACUUM BKR VB-10A 0901
VACUUM BKR VB-10B 0902
10
CONTAINMENT PURGE RBV-2 1006
CONTAINMENT VENT RBV-3 1007
11
RCDT GAS ANZR MG(R)-503 1101
RCDT VENT MG(R)-509 1102
RCDT PUMPS RC-507 1103
CONTAINMENT SUMP PUMP MD(R)-134 1104
R11/12 SAMPLE ISOL AS-1 1105
CONTAINMENT PURGE RBV-1 1106
CONTAINMENT VENT RBV-4 1107
12
RCDT GAS ANZR MG(R)-504 1201
RCDT VENT MG(R)-510 1202
RCDT PUMPS RC-508 1203
CONTAINMENT SUMP PUMP MD(R)-135 1204
R11/12 SAMPLE ISOL AS-2 1205
CONTAINMENT PURGE TAV-12 1206
CONTAINMENT VENT RBV-5 1207
11

B


CNTMT ISOL ACTIVE
1
2
3
4
5
6
7
LD ORIF 1A LD-4A 0101
LD ORIF 1B LD-4B 0102
LD ORIF 1C LD-4C 0103
DDT VENT WG-311 0106
DDT ISOLATION MD(R)-323A 0107
LD HX FLOW LD-6 0202
EXCS LD HX CC-653 0204
VCT VENT CVC-54 0205
DDT VENT WG-310 0206
DDT ISOLATION MD(R)-323B 0207
S/G 1A BD BT-2A 0301
S/G 1B BD BT-2B 0302
S/G 1A SAMPLE BT-31A 0303
S/G 1B SAMPLE BT-31B 0304
SEAL WATER LEAKOFF CVC-211 0307
S/G 1A BD BT-3A 0401
S/G 1B BD BT-3B 0402
S/G 1A SAMPLE BT-32A 0403
S/G 1B SAMPLE BT-32B 0404
ACMTR N2 SPLY NG-107 0406
SEAL WATER LEAKOFF CVC-212 0407
PRZR STM SMPL RC-402 0501
PRZR LIQ SMPL RC-412 0502
RCS HOTLEG B SAMPLE VALVE RC-422 0503
PRZR STM SMPL RC-403 0601
PRZR LIQ SMPL RC-413 0602
RCS HOTLEG B SAMPLE VALVE RC-423 0603
PRT GAS ANZR MG(R)-512 0604
PRT GAS ANZR MG(R)-513 0605
PRT N2 SPLY NG-302 0606
PRT MAKE-UP MU1010-1 0607
H2 VENT ISOL LOCA 2B 0701
H2 RCMBR TO CNTMT LOCA 201B 0702
H2 RCMBR LOCA 100B 0801
H2 DILUTE SA7003B 0802
R-11/12 SAMPLE ISOL AS-32 0805
CNTMT SPRAY TEST LINE ICS-201 0806
CNTMT SPRAY TEST LINE ICS-202 0807
VACUUM BKR VB-10A 0901
VACUUM BKR VB-10B 0902
CONTAINMENT PURGE RBV-2 1006
CONTAINMENT VENT RBV-3 1007
RCDT GAS ANZR MG(R)-503 1101
RCDT VENT MG(R)-509 1102
RCDT PUMPS RC-507 1103
CONTAINMENT SUMP PUMP MD(R)-134 1104
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CONTAINMENT PURGE RBV-1 1106
CONTAINMENT VENT RBV-4 1107
RCDT GAS ANZR MG(R)-504 1201
RCDT VENT MG(R)-510 1202
RCDT PUMPS RC-508 1203
CONTAINMENT SUMP PUMP MD(R)-135 1204
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CONTAINMENT VENT RBV-5 1207
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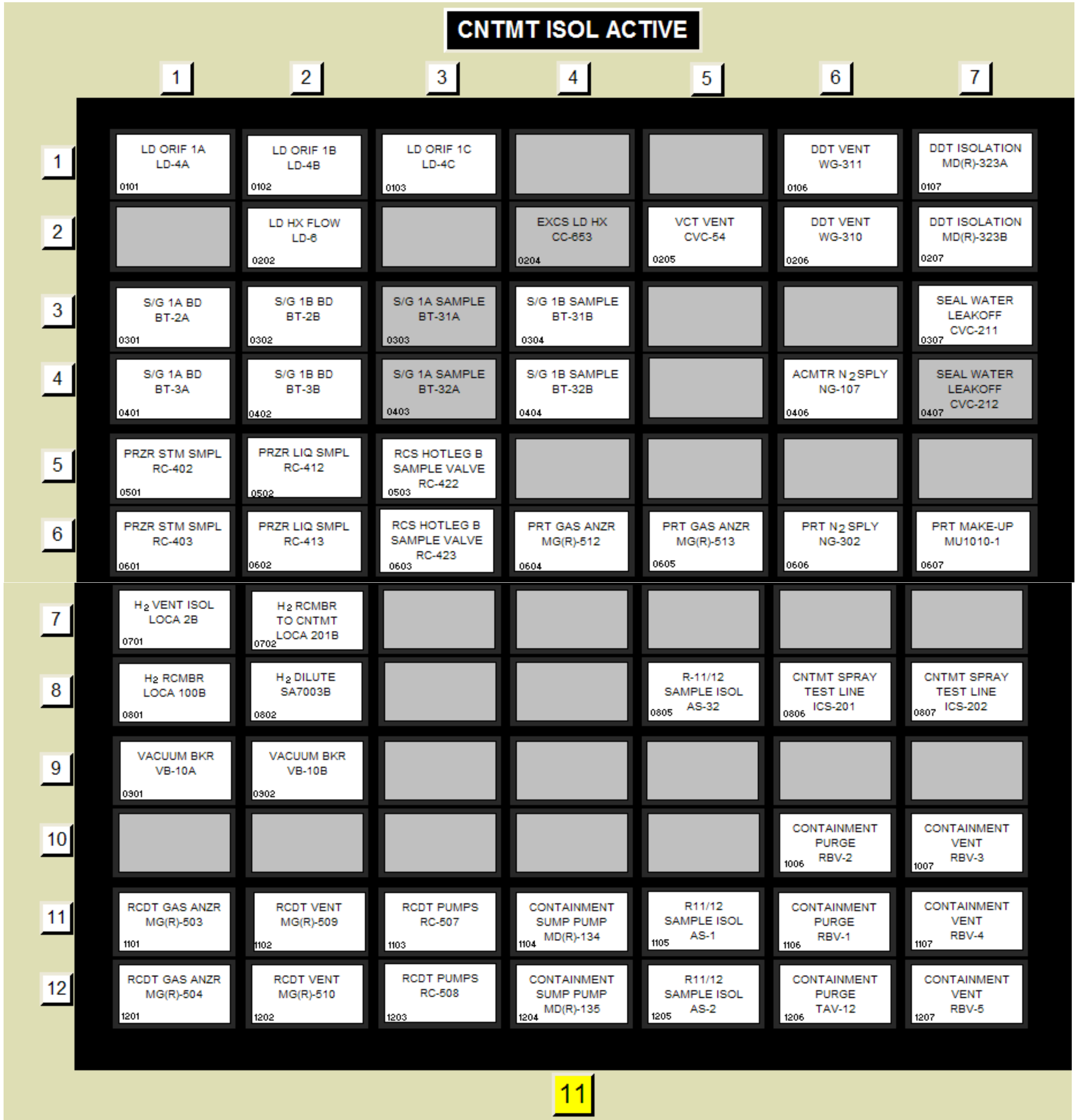
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DDT VENT WG-311 0106
DDT ISOLATION MD(R)-323A 0107
LD HX FLOW LD-6 0202
EXCS LD HX CC-653 0204
VCT VENT CVC-54 0205
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S/G 1A BD BT-3A 0401
S/G 1B BD BT-3B 0402
S/G 1A SAMPLE BT-32A 0403
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ACMTR N2 SPLY NG-107 0406
SEAL WATER LEAKOFF CVC-212 0407
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RCS HOTLEG B SAMPLE VALVE RC-422 0503
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PRZR LIQ SMPL RC-413 0602
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PRT GAS ANZR MG(R)-512 0604
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PRT MAKE-UP MU1010-1 0607
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H2 RCMBR TO CNTMT LOCA 201B 0702
H2 RCMBR LOCA 100B 0801
H2 DILUTE SA7003B 0802
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CNTMT SPRAY TEST LINE ICS-202 0807
VACUUM BKR VB-10A 0901
VACUUM BKR VB-10B 0902
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CONTAINMENT VENT RBV-3 1007
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RCDT PUMPS RC-507 1103
CONTAINMENT SUMP PUMP MD(R)-134 1104
R11/12 SAMPLE ISOL AS-1 1105
CONTAINMENT PURGE RBV-1 1106
CONTAINMENT VENT RBV-4 1107
RCDT GAS ANZR MG(R)-504 1201
RCDT VENT MG(R)-510 1202
RCDT PUMPS RC-508 1203
CONTAINMENT SUMP PUMP MD(R)-135 1204
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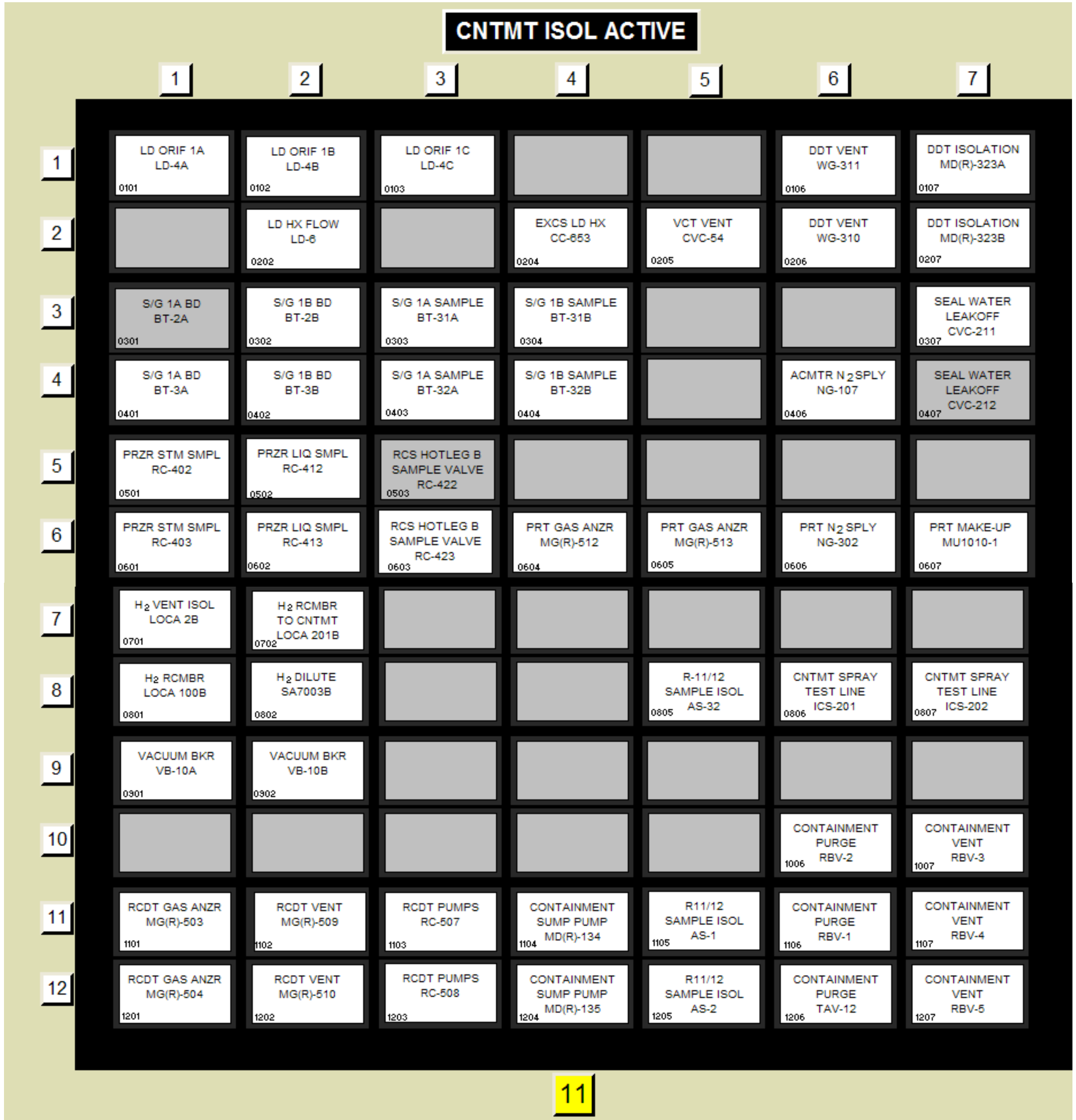
D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DDT VENT WG-311 0106
DDT ISOLATION MD(R)-323A 0107
LD HX FLOW LD-6 0202
EXCS LD HX CC-653 0204
VCT VENT CVC-54 0205
DDT VENT WG-310 0206
DDT ISOLATION MD(R)-323B 0207
S/G 1A BD BT-2A 0301
S/G 1B BD BT-2B 0302
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S/G 1B SAMPLE BT-31B 0304
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S/G 1A BD BT-3A 0401
S/G 1B BD BT-3B 0402
S/G 1A SAMPLE BT-32A 0403
S/G 1B SAMPLE BT-32B 0404
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PRZR LIQ SMPL RC-413 0602
RCS HOTLEG B SAMPLE VALVE RC-423 0603
PRT GAS ANZR MG(R)-512 0604
PRT GAS ANZR MG(R)-513 0605
PRT N2 SPLY NG-302 0606
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H2 VENT ISOL LOCA 2B 0701
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RCDT PUMPS RC-507 1103
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CONTAINMENT PURGE RBV-1 1106
CONTAINMENT VENT RBV-4 1107
RCDT GAS ANZR MG(R)-504 1201
RCDT VENT MG(R)-510 1202
RCDT PUMPS RC-508 1203
CONTAINMENT SUMP PUMP MD(R)-135 1204
R11/12 SAMPLE ISOL AS-2 1205
CONTAINMENT PURGE TAV-12 1206
CONTAINMENT VENT RBV-5 1207
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33. Given the following:

- The Plant is in MODE 3 when a Design Basis LOCA occurs
- The crew has transitioned to E-1, “Loss of Reactor or Secondary Coolant”
- RWST level is 55% and slowly lowering
- The STA informs the Unit Supervisor of an ORANGE path for CONTAINMENT due to BOTH Internal Containment Spray pumps tripping on overcurrent
- The crew transitions and COMPLETES FR-Z.1, "Response to High Containment Pressure"
- The ORANGE path for CONTAINMENT still exists
- NO other ORANGE or RED paths exists

Which of the following states the expected response of Containment Pressure at the completion of FR-Z.1?

Containment pressure will . . .

A. not exceed design pressure as long as four Containment Fan Coil Units are operating to provide heat removal for the steam produced from the Design Basis LOCA.

B. continue to rise until an equilibrium is established between steam production from the Design Basis LOCA and the rate of heat removal provided by the containment dome fans.

C. slowly lower because the combination of one train of Containment Fan Coil Units and Shield Building Ventilation are sufficient to lower containment pressure during a Design Basis LOCA.

D. exceed Containment Design pressure unless one Internal Containment Spray Pump can be started, the heat removal capacity of the containment fan coil units and containment dome fans are not sufficient during a Design Basis LOCA.

Answer: A

Cognitive Level:

Comprehension 2-DR: The operator recognizes the relationship between ECCS (specifically ICS pumps, CFCU and containment design pressure limits) while controlling ECCS equipment per procedure guidance.

K/A:

006A1.07 – Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ECCS controls including: Pressure, high and low RO/SRO IMP 3.3/3.6

Objective:

RO2-01-LP023.002 – DESCRIBE the function(s)/purpose(s), design basis, and operating characteristics of the ICS System.

Reference:

USAR Section 6.4.1

FR-Z.1, Response to High Containment Pressure

BKG-FR-Z.1, Background Response to High Containment Pressure

Source:

New

Justification:

A CORRECT	Four fan can units are sufficient to prevent containment pressure from exceeding design pressure during a design basis LOCA.
B INCORRECT	Containment Dome fans are not part of the design analysis for containment pressure during a design basis LOCA.
C INCORRECT	Need both trains of containment fan coil units during a design basis LOCA to protect containment and lower pressure. Shield building ventilation is not considered in the analysis for containment pressure during a design basis LOCA.
D INCORRECT	As Long as the four containment fan coil units are operating then containment design pressure will not be exceeded.

34. Given the following:

- The unit is at 100% Rated Thermal Power.
- Pressurizer Pressure is within normal operating range for operation at 100% Rated Thermal Power when PR-2B, Pressurizer PORV, fails 40% OPEN.

What are the Immediate Operator Actions in the correct order of performance <u>AND</u> the expected PRT pressure response of PR-2B failing OPEN 40%?

(Immediate Operator Actions are completed within 1 minute)

A. 1. CHECK Pressurizer Pressure < 2315 psig
   2. CLOSE PR-2B
   3. CLOSE Pressurizer Block Valve PR-1B
   <u>AND</u>
   PRT pressure will remain less than 200 psig due to the quench volume of the PRT.

B. 1. CHECK Pressurizer Pressure < 2335 psig
   2. CLOSE Pressurizer Block Valve PR-1B
   3. CLOSE PR-2B
   <u>AND</u>
   PRT pressure will equalize with containment pressure due to the PRT rupture disk rupturing when PRT pressure reaches 120 psig.

C. 1. CHECK Pressurizer Pressure < 2335 psig
   2. CLOSE PR-2B
   3. CLOSE Pressurizer Block Valve PR-1B
   <u>AND</u>
   PRT pressure will equalize with containment pressure when the piping rupture disk ruptures at 234 psig.

D. 1. CHECK Pressurizer Pressure < 2315 psig
   2. CLOSE Pressurizer Block Valve PR-1B
   3. CLOSE PR-2B
   <u>AND</u>
   PRT pressure will increase consistent with the volume discharged from the Pressurizer.

Answer: D

Cognitive Level:

Comprehension 2-RI: The operator must recognize the correct order of the procedure as well as the relationship between the RCS, PZR PORVs and PRT for the given conditions.

K/A:

007A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Stuck-open PORV or code safety. RO/SRO IMP 3.9/4.2

Objective:

RO-02-01-LP26B.001 - Describe the function/purpose, design basis and operating characteristics of the PRZR and PRT systems.

Reference:

OP-AP-104, Emergency and abnormal Operating Performance, step 3.5.5
AOP-GEN-001, Operator Immediate Actions.
OPERXK-100-10, Flow Diagram – Reactor Coolant System

Source:

New

Justification:

A INCORRECT	The immediate actions are performed in the order of closing PR-1B first then PR-2B. OP-AP-104 step 3.5.5 states that immediate operator actions are performed in the proper sequence.
B INCORRECT	The initial pressure is 2315 psig not 2335 psig. 2335 psig is the pressure at which the PORV will open. The PRT rupture disk will not reach its setpoint of 100 psig with initial conditions in the PRT normal.
C INCORRECT	The initial pressure is 2315 psig not 2335 psig. The piping rupture disk is not expected to rupture.
D CORRECT	The correct values and sequence of immediate actions. PRT response is correct.

35. Given the following:

- The unit was operating at 100% Rated Thermal Power when a reactor trip and Safety Injection occurred.
- Bkr 15201, Bus 52 Supply Bkr, OPENED on the reactor trip
- Diesel Generator 'A' failed to start.

With NO Operator action, which of the following equipment conditions is correct?

	Running	NOT Running
A.	ASV Exhaust Fan ‘B’	RHR Pump ‘A’
B.	Component Cooling pump ‘A’	Charging Pump ‘B’
C.	Safety Injection pump 'B'	Turbine Driven AFW Pump
D.	Service Water Pump 1A1	CRPA Recirculation Fan ‘B’

Answer: B

Cognitive Level:

Comprehension 2-DR: Using the plant conditions given, demonstrate an understanding between the relationship of DG operation, Safety Injection and power supplies.

K/A:

008K2.02 - Component Cooling Water. Knowledge of bus power supplies to the following: CCW Pump, including emergency backup. RO/SRO IMP 3.0/3.2

NOTE: Kewaunee does not have an emergency backup bus power supply for CCW.

Objective:

RO2-01-LP031.002 - Describe the Component Cooling Water System to include the following in the description.

3. Interfaces with the following:

480 VAC Electrical Distribution System.

References:

E-914, Interlock Logic Diagrams – Bus 1-5 Source Breakers
E-1634, Integrated Logic Diagrams – Diesel Generator Electric
E-240, Circuit Diagram – 4160V and 480V Power Sources
E-844, Circuit Diagram – 125VDC
NCL-CVC-001, Charging and Volume Control Prestartup Checklist
N-CC-31-CL, Component Cooling System Prestartup Checklist
E-2025, Integrated Logic Diagrams – Chemical & Volume Control System

Source:

New

Justification

A	INCORRECT	RHR pump will be automatically started by the SI signal and is powered from bus 5.
B	CORRECT	Charging Pump B will not be running, stopped by the SI signal, it does have power. Component Cooling pump 'A' would get a start signal from the SI signal and is powered from Bus 51 not bus 52.
C	INCORRECT	SI Pump 'B' would be running. The TDAFW pump would also be running, since the auxiliary lube oil pump is powered from DC (BRA-104), this would not cause the pump not to start.
D	INCORRECT	SW pump 1A1 is powered from bus 5 and CPRA fan will start on SI.

36. Given the following:

- The unit is at 100% Rated Thermal Power.
- The following indications are present in the Component Cooling (CC) Water System:
    - CC Pump 'A' - running
    - CC Pump 'B' - standby
    - CC return flow from the RXCP 'A' - 245 gpm and stable
    - Radiation Monitor R-17, Component Cooling, in ALERT
    - CC Surge tank level - 55% and slowly rising
    - CC header pressure - 32 psig and slowly lowering

What is an expected plant response for the current conditions?

A. The running Component Cooling Pump, CC Pump 'A', receives a TRIP signal.

B. The standby Component Cooling Pump, CC Pump 'B', receives an auto START signal.

C. CC-610A, RXCP Thermal Barrier CC Isolation Valve, receives an automatic signal to CLOSE.

D. CC-104, Component Cooling Surge Tank Vent Valve receives an automatic signal to CLOSE.

Answer: B

Cognitive Level:
Memory - 1-I: Recall the system response to various signals in the CC system.

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

Objective:
RO2-01-LP031.004 - Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Component Cooling Water System and the following major system components:
- Component Cooling Water Surge Tank
- Component Cooling Water Pumps A and B
- RXCP A/B Thermal Barr Comp Cooling Return (CC-610A and 610B)

Reference:
ARP-47021-H, CC Pumps Disch Pressure Low
AOP-RM-001, Abnormal Radiation Monitoring System, Attachment A, step A8
ARP-47024-H, CC Surge Tank Level High/Low
ARP-47014-J, RXCP Therm Barrier CC Temp High, Comments Section
AOP-CC-001, Abnormal Component Cooling Operation
E-2045, Integrated Logic Diagram – Component Cooling System

Source:
New

Justification:

A INCORRECT	It is common to generate a trip signal to the running CC pump on lowering surge tank level. The alarm for low surge tank level actuates at 45%. However per ARP 47024-H there are no automatic actions on surge tank level lowering.
B CORRECT	Per 47021-H, The standby CC pump automatically starts at 35 psig.
C INCORRECT	Per ARP 47014-J, RXCP Thermal Barrier Component Cooling Return Valves automatically close on high thermal barrier return flow of 260 gpm. The setpoint for automatic closure has not been exceeded.
D INCORRECT	It is common to generate a closed signal on the CC surge tank vent valves on a high rad signal in the CC system. However, Per AOP-RM-001, R-17, Comp Cooling Liquid Monitor Automatic Actions: None.

37. Given the following:

- The unit was operating at 60% power as indicated on Nuclear Instruments (NI), and stable.
- All Pressurizer Backup Heaters are in AUTO
- Charging Pump 'A' is operating in AUTO
- Charging Pump 'B' is operating in MANUAL
- A transient occurred with the following results:

Parameter	Value	Trend
Reactor Power (NI)	60%	Stable
Pressurizer Level	48%	Rising Slowly
Tave	562ºF	Rising Slowly
Pressurizer Pressure	2263 psig	Rising Slowly

Which of the following combinations of listed pressurizer heaters and spray is the correct response for the current status?

	Pressurizer Spray Valves PS-1A & PS-1A	Backup Heaters	Heater Group 'C'
A.	CLOSED	ON	MAX POWER
B.	THROTTLED	ON	MIN POWER
C.	CLOSED	OFF	MIN POWER
D.	THROTTLED	OFF	MAX POWER

Answer: B

Cognitive Level:

Analysis 3-PEO: Requires the student to recall the Pressurizer response to the given conditions for pressure and level. The student must then apply this knowledge to recognize the configuration of each Heater Group A, B, D and E, variable Heater Group C, and Pressurizer spray.

K/A:

010A4.02 - Ability to manually operate and/or monitor in the control room: PZR heaters. RO/SRO IMP 3.6/3.4

Objective:

R02-05-LP36C.004 - DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls, and alarms associated with the Pressurizer Pressure Control System and the following major system components.
Pressurizer Heaters

References:

E-2038, Integrated Logic Diagram – Reactor Coolant System
E-2039, Integrated Logic Diagram – Reactor Coolant System
ARP-47043-E, Pressurizer Level Deviation

Source:

Master Bank 004K3.07 02 - modified
Bank 2009 Audit Exam # 24

Justification:

At 60% Power, Programmed Pressurizer level setpoint is 36.4%. Heater Groups ABDE will be on due to a greater than 10 % Pressurizer level change (11.58% level change.) Heater Group C will be at Min due to Pressurizer pressure of 2260. Variable Heater Group C operates at Max power at 2220 and Min at 2250. Pressurizer sprays throttle open at 2260 and are full open at 2310.

A INCORRECT Heat Group C will be at Min power due to a pressure of 2260. and Spray will be throttled open.

B CORRECT See above.

C INCORRECT Backup Heaters would be on and the spray valves would be throttled open.

D INCORRECT Backup Heaters would be on and Heater Group C would be at min power.

38. Given the following:

- The unit is operating at 100% Rated Thermal Power.
- A total loss of main feedwater occurs.
- A reactor trip signal is generated, but the Reactor Trip Breakers FAIL to OPEN.
- A Rod Control Urgent Failure prevents rod motion.
- Auxiliary Feedwater System is operating per design.
- Operators have been dispatched to locally open the Reactor Trip Breakers.

What immediate action should be taken to mitigate this transient <u>AND</u> what is the purpose of the immediate action?

	Immediate Action	Purpose
A.	Align maximum Auxiliary Feedwater flow to ONE Steam Generator	Preserve a heat sink for RCS heat removal
B.	Manually trip the Main Turbine	Preserve the water inventory in the Steam Generators
C.	Open the Pressurizer PORVs	Avoid an RCS over-pressurization as the Pressurizer goes water solid
D.	Slowly reduce turbine load	Prevent the PRZR PORVs from lifting by avoiding a rapid RCS temperature and pressure rise

Answer: B

Cognitive Level:
Memory 1-B: Recall steps and purpose of immediate actions of FR-S.1.
K/A:
012G2.4.49 - Reactor Protection: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. RO/SRO IMP 4.6/4.4
Objective:
RO4-04-LP008.003 - Given a set of plant conditions, recommend the appropriate procedural actions to be taken while implementing FR-S.1, Response to Nuclear Power Generation/ ATWS.
Reference:
E-0, Reactor Trip or Safety Injection, step 1
FR-S.1Response to Loss of Core Shutdown
BKG-FR-S.1, Background for Response to Loss of Core Shutdown, step 2
Source:
Bank - INPO bank -Point Beach 1 9/29/2003
2006 NRC Retake Exam

Justification:

Per E-0, step 1, based on the given conditions, transition to FR-S.1. The immediate actions of step FR-S.1 are to trip "Verify Reactor Trip." Based on the given conditions, actions are complete for step 1 with an operator being dispatched to open the reactor trip and bypass breakers. Step 2 of FR-S.1 is "Verify Turbine Trip". The RNO is to "Manually trip the turbine" background for FR-S.1 states, "For an ATWS event where a loss of normal feedwater has occurred, analyses have shown that a turbine trip is necessary (within 30 seconds) to maintain SG inventory."

A INCORRECT Stopping power generation and preserving SG inventory are more effective strategies.

B CORRECT This preserves SG water inventory as long as possible.

C INCORRECT The most important priority is Turbine trip and SG water inventory.

D INCORRECT RCS pressure increase is a function of SG inventory loss, not mitigated by Pressurizer PORVs.

39. Given the following:

- The unit was at 100% Rated Thermal Power.
- An equipment failure causes an inadvertent Safety Injection (SI) Signal.
- The operator determines the SI Signal is invalid and depresses the SI RESET pushbuttons 15 seconds after the SI initiation.

Which of the following is the result of the above operator actions?

A. The running charging pumps will remain operating.

B. SI RESET pushbuttons will need to be depressed again to reset SI.

C. Containment Isolation will NOT occur due to resetting the SI signal.

D. ALL components normally started by the SI signal will NOT automatically start.

Answer: B

Cognitive Level:

Memory 1-I: Recall the requirements to reset an ESFAS SI signal. Determine the effect of premature reset of an SI signal.

K/A:

013K1.18 - Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: Premature reset of ESF actuation.

Objective:

RO2-05-LP033.004 - DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Safety Injection System and the following major system components:

1. Refueling Water Storage Tank
2. Safety Injection Pumps
3. Safety Injection Accumulators
4. Motor Operated Valves
   a. SI-4A(B), Refueling Water Storage Tank to SI Pump
   b. SI-5A(B), Safety Injection Pump 1A(B) Suction Isolation
   c. SI-9A, Safety Injection Cold Leg Isolation
   d. SI-9B, Safety Injection to Reactor Vessel Isolation
   e. SI-11A(B), Safety Injection Loop A(B) Cold Leg Isolation
   f. SI-15A(B), Reactor Vessel Safety Injection Isolation
   g. SI-20A(B), SI Accumulator 1A(B) Discharge Isolation
   h. SI-208A(B), Refueling Water Storage Tank Test Inlet Stop
   i. SI-300A(B), RWST Supply to RHR Pump A(B)
   j. SI-302A(B), RHR Pump A(B) Injection to Reactor Vessel
   k. SI-350A(B), Containment Sump B Isolation
   l. SI-351A(B), Containment Sump B Isolation

RO2-05-LP055.004 - Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Engineering Safety Features System and the following major system components: ESF Actuation.

References:

XK-100-150, Logic Diagrams – Safeguards Actuation Signals, Rev. 5F
E-1635, Integrated Logic Diagram – Diesel Generator Electric, Rev. O

Source:

KNPP Initial Bank - Modified Stem and three choices.
Old Kewaunee Bank - Question 0330000001K0401

Justification:

A INCORRECT	The signal will complete the sequence and the charging pumps will be stopped.
B CORRECT	The 90 second timer needs to time out before SI can be reset. Therefore the SI Reset pushbuttons will have to be depressed again to reset SI.
C INCORRECT	Containment Isolation occurs when a SI signal is received. Resetting SI does not affect Containment Isolation activation.
D INCORRECT	All Components normally started by the SI signal will start. The reset pushbuttons will have no affect on the starting of components.

40. Given the following:

- The Unit was operating at 100% Rated Thermal Power when the unit experienced a spurious reactor trip.
- During the resulting transient a Bus 3 and Bus 5 LOCKOUT occurred.

With NO operator action, which of the following lists equipment that would be RUNNING two (2) minutes after the transient?

A. RXCP ‘B’, Containment Fan Coil Unit ‘D’, & Reactor Make Up Pump ‘B’

B. Rod Drive MG Set 1B, Main Feed Pump ‘A’, & Heater Drain Pump ‘A’

C. Boric Acid Transfer Pump ‘B’, Rod Drive MG Set 1A, & Fire Pump ‘B’

D. Containment Fan Coil Unit ‘B’, Containment Dome Fan ‘B’, & Condensate Pump ‘B’

Answer: A

Cognitive Level:
Comprehension 2-DR: Recognize the relationship between running equipment and the loss of buses.

K/A:
022K2.01 - Knowledge of bus power supplies to the following: Containment cooling fans. RO/SRO IMP 3.0/3.1

Objective:
Various for Power supplies of Equipment

References:
E-2020, Integrated Logic Diagram – Make Up Water System Sh1
E-240, Circuit Diagram – 4160V & 480V Power Sources
E-244, Circuit Diagram – Generator & 4160V Equipment
E-235, Circuit Diagram – 480V SWGR Safeguard Buses

Source:
New

Justification:

A CORRECT — RXCP 'B' running, CFCU 'D' powered from Bus 61, running. Reactor Makeup transfer pump 'B' would turn on when 'A' turns off on loss of the bus 3, pressure switch for Auto start at 80 psig in the header.

B INCORRECT — Rod Drive MG Set 1B running, Main Feedwater Pump 'A' running (does not trip on reactor trip); Heater Drain Pump 'A' is powered from Bus 3 therefore is off.

C INCORRECT — Boric Acid Transfer Pump would not be running unless the operator turns on the pump when doing a boration. The Jockey Pump will remain running and will maintain fire header pressure, therefore Fire Pump 'B' will not start. Rod Drive MG Set 1A would be remain running as the cross tie closes.

D INCORRECT — CFCU 'B' does not have power as it is powered from Bus 51 which is de-energized. Containment Dome Fan 'B' is powered from MCC-62B and Condensate Pump 'B' is powered from bus 4, so both are running.

41. Given the following:

12:00:00 – The unit is in MODE 1 with the Shroud Cooling Coil ‘A/B’ in service
12:00:02 – Safety Injection (SI) occurs
12:00:04 – The power supply to the Train ‘A’ Service Water (SW) Valves for Shroud Cooling fails, no other components are affected by the power loss.

Which of the following describes the expected valve response and indication of SW flow to Containment Fan Coil Unit (CFCU) ‘B’?

‘Shroud Cooling Coil A/B’ SW valves . . .

A. are unaffected by the SI Signal and continue to REGULATE SW flow to CFCU ‘B’.

B. reposition to isolate SW flow to Shroud Cooling Coil ‘A/B’ resulting in a RISE in SW flow to CFCU ‘B’.

C. reposition to LIMIT the combined SW flow through Shroud Cooling Coil ‘A/B’ and CFCU ‘B’ to 50 gpm.

D. remain in the same position and a CONSTANT SW flow rate is maintained to Shroud Cooling Coil ‘A/B’ and CFCU ‘B’.

Answer: B

Cognitive Level:

Comprehension 2-RI: The operator recognizes the interaction between the Safety Injection Actuation and the Service Water to Shroud Cooling valve given a loss of power to one of the Trains

K/A:

022K3.02 – Knowledge of the effect that a loss or malfunction of the CCS will have on the following: Containment instrumentation readings. RO/SRO IMP 3.0/3.3

Objective:

RO2-04-LP018.002 – DESCRIBE the Reactor Building Ventilation System; include the following in the description:

- Major system flow paths
- Function/purpose, design basis, operating characteristics, and physical locations appropriate for the following major components:
- Interfaces with the following plant systems:
  a. Service Water System

Reference:

E-3218, S/D Shroud Cooling SW Isolation Valves CV-31694, CV-31695, CV-31696, CV-31697, CV-31698, CV-31704, and CV-31705
E-3174, Integrated Logic Diagram – Shroud Cooling System Modification
SDBD-KPS-SW, System Design Basis Document for Service Water System, Table 4.1-1 and 6.3-1
OPERM-547, Flow Diagram – Service Water System Containment Cooling

Source:

New

Justification:

A INCORRECT — The valves DO receive an SI signal. The SW flow does NOT remain constant, the valves reposition to a fail position maximizing the flow of SW to the ‘B’ CFCU.

B CORRECT — The valves DO reposition on the loss of power and go to a fail position isolating the shroud cooler, which DOES maximize the SW flow to the ‘B’ CFCU.

C INCORRECT — The valves do NOT reposition to allow 50 gpm flow to the shroud cooling coils.

D INCORRECT — The valves did lose power, therefore they do NOT remain in the same position, they reposition to a fail position which DOES NOT maintain constant SW flow to the ‘B’ CFCU.

42. Given the following:

- The unit is in MODE 3.
- SP-23-100A, “Train A Containment Spray Pump and Valve Test – IST”, is in progress.
- After starting Internal Containment Spray (ICS) Pump ‘A’, but before aligning ICS train ‘A’ for the full flow test, a Design Basis LOCA occurs.
- The crew enters the EOP network.
- Containment pressure is 25 psig and rising at the completion of immediate actions.

What is the status of the Internal Containment Spray Train ‘A’?

(SP-23-100A is provided as a reference)

ICS pump 'A' is providing . . .

A. flow to ICS ring header ‘A’.

B. recirculation flow to the suction of the pump.

C. recirculation flow to the Refueling Water Storage Tank.

D. no flow through the system and operating at shutoff head.

Answer: D

Provided Reference:

SP-23-100A, "Train A Containment Spray Pump and Valve Test – IST"

Cognitive Level:

Analysis 3-SPR: Recognizing the relationship between containment spray and containment cooling. Given ICS-7A operation, determine how containment spray will be effected.

K/A:

026K3.01- Knowledge of the effect that a loss or malfunction of the CSS will have on the following: CCS. RO/SRO IMP 3.9/4.1

Objective:

RO2-01-LP023.004 - Describe the operation, automatic actions, interlocks, instrumentation, controls, and alarms associated with the ICS System and the following major system components. Containment Spray Pumps, Spray Nozzles and Rings.

RO2-01-LP023.002 - Describe the ICS system, Include the following in the description, function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components.

Reference:

OPERM-217, Operation Flow Diagrams – Internal Containment Spray System

SP-23-100A, Train A Containment Spray Pump and Valve Test - IST

Source:

New

Justification:

A INCORRECT ICS-7A does not receive an OPEN signal with ICS, therefore since it was closed during the SP spray flow will not be provided to ring 'A'.

B INCORRECT Normal Recirculation flow goes to the RWST.

C INCORRECT Recirculation flow to the RWST is isolated.

D CORRECT Pump is operating at shutoff head.

43. Given the following:

- The unit was operating at 100% Rated Thermal Power when a leak in the Main Steam System occurred.
- Annunciator 47052-F, STM Exclusion Area Temp High, is LIT.
- Annunciator 47051-F, STM Exclusion Area Isolation, is LIT.

Which of the following is an expected plant effect?

A. Zone SV boundary dampers close and Zone SV Exhaust fans start.

B. SI Active Status Panel Light 44901-0901, Zone SV EXH Fan A, is dim.

C. Control Room A/C Fans and Auxiliary Building Supply Air Vent Fans trip.

D. Zone SV Exhaust fan discharge dampers close and Zone SV Exhaust fans trip.

Answer: A

Cognitive Level:
Memory 1-I: Recall the plant response to a Steam Exclusion Area Isolation and actuations that provide for Reactor Building Isolation.

K/A:
039K4.07: Knowledge of MRSS design feature(s) and/or interlock(s) which provide for following: Reactor Building Isolation. RO/SRO IMP 3.4/3.7

Objective:
RO2-04-LP014.004 - Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Auxiliary Building Special Ventilation and the following major system components:

Zone SV Exhaust Filter Assemblies
Deluge System
Zone SV Exhaust Fans
Zone SV Filter Assembly Inlet Dampers
Zone SV Filter Assembly Outlet Dampers
Relief Backdraft Dampers
Boundary Dampers
Zone SV Boundary Doors
Steam Exclusion Dampers

Reference:
ARP-47052-F, STM Exclusion Area Temp High
ARP-47051-F, STM Exclusion Area Isolation
E-2074, Integrated Logic Diagram – Aux Bldg Special Vent
E-1616, Integrated Logic Diagram – Aux Bldg Spec Vent
E-2608, Schematic Diagram – Steam Exclusion Annunciators
OPERM-601, Operation Flow Diagram – Turbine & Aux Bldg Ventilation

Source:
New

Justification:

ASV system start and Zone SV boundary isolation initiated by any of the following:

- SI initiation
- Aux Bldg Vent Stack High Radiation
- Steam Exclusion Zone SV isolation.

The following actions occur upon ASV system start:

- Zone SV Boundary Dampers close
- Zone SV Exhaust Filter Inlet Dampers open
- Zone SV Exhaust Fan Discharge Dampers open
- Zone SV Exhaust Fans start
- Zone SV Exhaust Filter Assembly Heating Coils energize
- Aux. Bldg. Supply Fans trip off
- Aux. Bldg. Exhaust Fans trip off
- Emergency Fan Coil Units start

A CORRECT On a Steam Exclusion Area Isolation, fans start and dampers close.

B INCORRECT If an SI Active Status Panel light is lit, the equipment is in the SI required state. In this case an SI will result in a Zone SV actuation, therefore the light will be lit on the active status panel for Zone SV Exhaust Fan A.

C INCORRECT The Aux Bldg Supply and Exhaust Fans will trip. However the Control Room Supply and Exhaust Fans will not.

D INCORRECT Zone SV Exhaust Fan Discharge Dampers open on the actuation, not close. Zone SV Exhaust Fans will start.

44. Given the following:

- The unit is at 85% Rated Thermal Power.
- A Feedwater (FW) control malfunction results in the following:
  - S/G A level: 39% and stable
  - S/G B level: 72% and rising

Which of the following is a result of the above conditions?

A. Only FW Pump 'A' is OPERATING

AND

FW Isolation Valve, FW-12B is CLOSED

B. BOTH FW Pumps are OPERATING

AND

FW Regulating Valve, FW-7B is CLOSED

C. Both FW Pumps are TRIPPED

AND

FW Isolation Valve, FW-12B is CLOSED

D. FW Regulating Valve FW-7B is CLOSED

AND

BOTH FW Isolation Valves FW-12A/B are CLOSED

Answer: C

Cognitive Level:

Memory 1-I: Recall the automatic actions associated with high level in one Steam Generator.

K/A:

059K4.19: Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: Automatic feedwater isolation of MFW. RO/SRO IMP 3.2/3.4

Objective:

RO2-02-LP05A.004 - Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Main Feedwater system and the following major components:

Main Feedwater Pumps
FW pumps lube oil system
FW pumps Recirculation Control Valves
FW pumps Discharge Valves (MOVs)
FW Bypass Control Valves
Main FW Control Valves
Main FW Isolation Valves (MOVs)

Reference:

ARP-47063-D, S/G B Feedwater Isolation

Source:

Bank - modified, Modified the stem and all 4 choices
KNPP Initial Bank RO2-02-LP05A.004 08

Justification:

S/G B Feedwater Isolation is caused by any of the following:

- A OR B Train Safety Injection signal.
- 2/3 channels B S/G levels greater than 67%
- Reactor Trip AND 2/4 channels Low Tave less than 554ºF.

If activated by S/G level greater than 67% then the following will occur:

- Turbine Trip
- Trips both FW pumps
- Closes FW-12B, Feedwater to Steam Generator 1B Isolation
- Closes FW-7B, S/G B Main AND FW-10B S/G B Bypass

A INCORRECT Both FW pumps will trip. FW-12B will close.

B INCORRECT Both FW pumps will trip and FW-7B will close.

C CORRECT Both FW pumps will trip. FW-12B will close.

D INCORRECT FW-7B will close. FW-12B will close. FW-12A will remain open.

45. Given the following:

- A reactor trip occurred from 100% Rated Thermal Power.
- All Auxiliary Feedwater (AFW) Pumps auto-started per design.
- A subsequent failure results in the loss of BRA-104, 125 VDC Bus, 30 seconds after Generator #1 Main Breaker G-1 OPENS.

Which of the following valve operations can successfully be performed from the Control Room?

A. MS-102, Turbine Driven AFW Pump Main Steam Isolation, can be used to stop the Turbine Driven AFW pump.

B. AFW-10A, AFW Train 'A' Crossover Valve, can be used to throttle discharge flow from the Turbine Driven AFW Pump.

C. SW-502, Service Water to Turbine Driven AFW Pump Isolation, can be aligned to the suction of the Turbine Driven AFW Pump.

D. AFW-10B, AFW Train 'B' Crossover Valve, can be used to isolate the Turbine Driven AFW Pump discharge flow to Steam Generator 'B'.

Answer: D

Cognitive Level:

Comprehension 2-RI: The student must recognize interaction between Auxiliary Feedwater and DC power to predict the results of a system actuation followed by a subsequent loss of power.

K/A:

061K6.01 - Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Controllers and Positioners.
RO/SRO IMP 2.5/2.8

Objective:

RO2-02-LP05B.003 - State the power supply/motive power for the following Auxiliary Feedwater System Components:

5. AFW-10A/AFW-10B
6. MS-100A, MS-100B
7. MS-102

References:

AOP-EDC-002, Loss of Train A Safeguards DC Power, Attachment A and Attachment B

Source:

New

Justification:

Per OP-KW-AOP-EDC-002, Attachment B, step B.1 The Auxiliary Feedwater System is affected as follows:

- MS-102 is de-energized (BRA-104 CKT 14)
- AFW-10A is de-energized (BRA-104 CKT 12)
- SW-502 is de-energized (BRA-104 CKT 15)

A INCORRECT MS-102 will not be able to be closed. The TDAFW Pump will not be able to be shut down.

B INCORRECT AFW-10A is not able to be throttled, it will fail as is.

C INCORRECT SW-502 will remain closed and cannot be re-aligned.

D CORRECT Train 'B' DC Power – BRB-104 ckt 32

46. Given the following: (Events Occur In Order Listed)

- A loss of Off-Site power occurred.
- Neither Diesel Generator (DG) automatically started.
- A reactor trip and Safety Injection then occurred.
- DG 'A' was started as directed by ECA 0.0, "Loss of All AC Power".
- DG 'A' is running and carrying load.
- DG 'A' load indicates 2900 KW.

Which of the following correctly describes the status of DG 'A'?

A. DG 'A' is within normal operational limits and may run continuously.

B. DG 'A' is rated for up to 2000 hours of operation at the current load.

C. Remove non-essential loads as necessary to reduce DG 'A' loading.

D. Additional loads on DG 'A' may be started as necessary up to 2979 KW.

Answer: C

Cognitive Level:
Memory 1-P: Recognize the current status of Diesel Generator loading and determine the correct action.

K/A:
062A1.01 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significance of D/G load limits.
RO/SRO Imp 3.4/3.8

Objective:
RO2-03-LP42A.001 - Describe the function/purpose, design basis, and operating characteristics of the Emergency Diesel Generator and the TSC Diesel Generator.

Reference:
NOP-DGM-001A, Diesel Generator A Remote Operation
ECA 0.0, Loss of All AC Power

Source:
New

Justification:

EDG Design:
- 2600 KW Continuous
- 2864 KW 2000 hr/yr
- 2950 KW 7 days/yr
- 3050 KW 30 min/yr

ECA 0.0, Check DG A load less than 2829KW. The RNO states, "Remove non-essential loads as necessary to reduce EDG loading less than 2829 KW."
OP-KW-NOP-DGM-001A Pre-requisites 4.6, “Do not exceed load of 2864 KW (upper end of 2000 hour rating).”

A INCORRECT The 2000 hr/year load rating is 2600 - 2864 KW.

B INCORRECT The load limit per ECA 0.0 is 2829 KW.

C. CORRECT Per ECA 0.0 and NOP-DGM-001A, Do not exceed 2829 KW. The required action is to remove non-essential loads as necessary to reduce EDG loading less than 2829 KW.

D INCORRECT The load rating which allows for continuous running is up to 2600 KW.

47. Given the following:

- The unit was operating at 100% Rated Thermal Power.
- A Station Blackout occurred.
- Station Battery 'A', BRA-101, is currently supplying plant DC loads.
- Discharge rate of Station Battery 'A', BRA-101, has been stable at 221 amp hours since the start of the Station Blackout.
- All station batteries were at their design capacity prior to the Station Blackout.

Which of the following is the operating design time of Station Battery ‘A’ at the current discharge rate?

A. 2 hours

B. 3 hours

C. 4 hours

D. 8 hours

Answer: D

Cognitive Level:
Memory1-F: Knowledge of Safeguards Battery Design Capacity

K/A:
063A1.01 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the D.C. Electrical System controls including: Battery capacity as it is affected by discharge rate. RO/SRO IMP 2.5/3.3

Objective:
RO2-03-LP038.002 - DESCRIBE the DC and Emergency AC Electrical Distribution System. Include the following in the description:

2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
   a. Batteries

Reference:
USAR Chapter 8 Section 8.2.3.4, Table 8.2-2

Source:
Bank: ILT: KNPP Initial Training Bank RO2-03-LP038.002
Changed the distracters to match values listed in the USAR.

Justification:

A INCORRECT 2 hours is less than the operating at the discharge rate of 221 AH.

B INCORRECT 3 hours design discharge rate is 464 AH.

C INCORRECT 4 hours corresponds to the SBO requirements.

D CORRECT 8 hours is the design operating time at 221 AH.

48. Given the following:

- Annunciator 47102-D, Instrument Bus Inverter Trouble, is LIT.
- SER printout identifies BRA-112 as the affected inverter.

With NO other faults, which of the following is a correct indication based on the above conditions?

If BRA-105 is supplying the inverter's load . . .

A. the inverter supplying load light would be ON.

B. the alternate source supplying load light would be ON.

C. annunciator 47101-A, BRA-102 DC Voltage Low, would be ON.

D. the red circuit status light on BRA-104 for BRA-112 supply breaker would be OFF.

Answer: B

Cognitive Level:
Memory 1-F: Knowledge of DC electrical system and indications

K/A:
063K4.01 - Knowledge of D.C. Electrical System design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control RO/SRO IMP 2.7/3.0

Objective:
RO2-03-LP38.004 - Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the DC and Emergency AC Electrical Distribution System and the following major system components:
5. Inverters

References:
E-233, Circuit Diagram – DC Aux and Emergency AC
ARP-47102-D, Instrument Bus Inverter Trouble
000-SD-38, System Description DC and Emergency AC Electrical Distribution System, 3.4.3, pages 16-17 and 3.11, page 21.

Source:
2009 ILT EXAM Question #47

Justification:

A INCORRECT — The normal supply for BRA-112 is MCC-52C.

B CORRECT — BRA-105 is the alternate source for BRA-112 and the alternate source supplying the load would be lit if it was supplying the inverter loads.

C INCORRECT — Annunciator 47101-A does not activate when BRA-105 is supplying BRA-112. BRA-112 is still energized, but on alternate power.

D INCORRECT — Indications for breaker status on BRA-104 does not indicate BRA-105 is supplying BRA-112. Red status light is an alarm module for that breaker, not an indication of power.

49. Given the following:

- Diesel Generator (DG) 'A' was started and is being prepared for parallel operations to the grid.
- See attached DG 'A' indications.

What will occur if Bkr 1-509, DG 'A' Output Breaker, closes under these conditions?

A. DG 'A' will PICK UP LOAD and the synchronizing lights will be OFF.

B. DG 'A' output CURRENT and SPEED will stabilize at a HIGHER value.

C. Bkr 1-509 will TRIP on reverse power and DG 'A' voltage will LOWER.

D. DG 'A' output voltage will RISE 4-6 volts and synchronizing lights will be ON.

Answer: A

Cognitive Level:

Analysis 3-PEO: Predict the outcome of closing the DG output breaker based on board indications of the grid and DG conditions.

K/A:

064A3.03 - Ability to monitor automatic operation of the ED/G system including: indicating lights, meters, and recorders. RO/SRO IMP 3.4/3.3

Objective:

RO2-03-LP42A.004 - Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Emergency Diesel Generator and TSC Diesel Generator Systems and the following major system components: DG Output Breaker.

Reference:

NOP-DGM-001A - Diesel Generator A Remote Operation
E-914, Interlock Logic Diagrams, Bus 1-5 Source Breakers

Source:

LORCycle00201 Bank
Question ID# 064A4.07, 0100040101A01
NOTE: Modified the stem and all the choices

Justification:

The provided indications are:

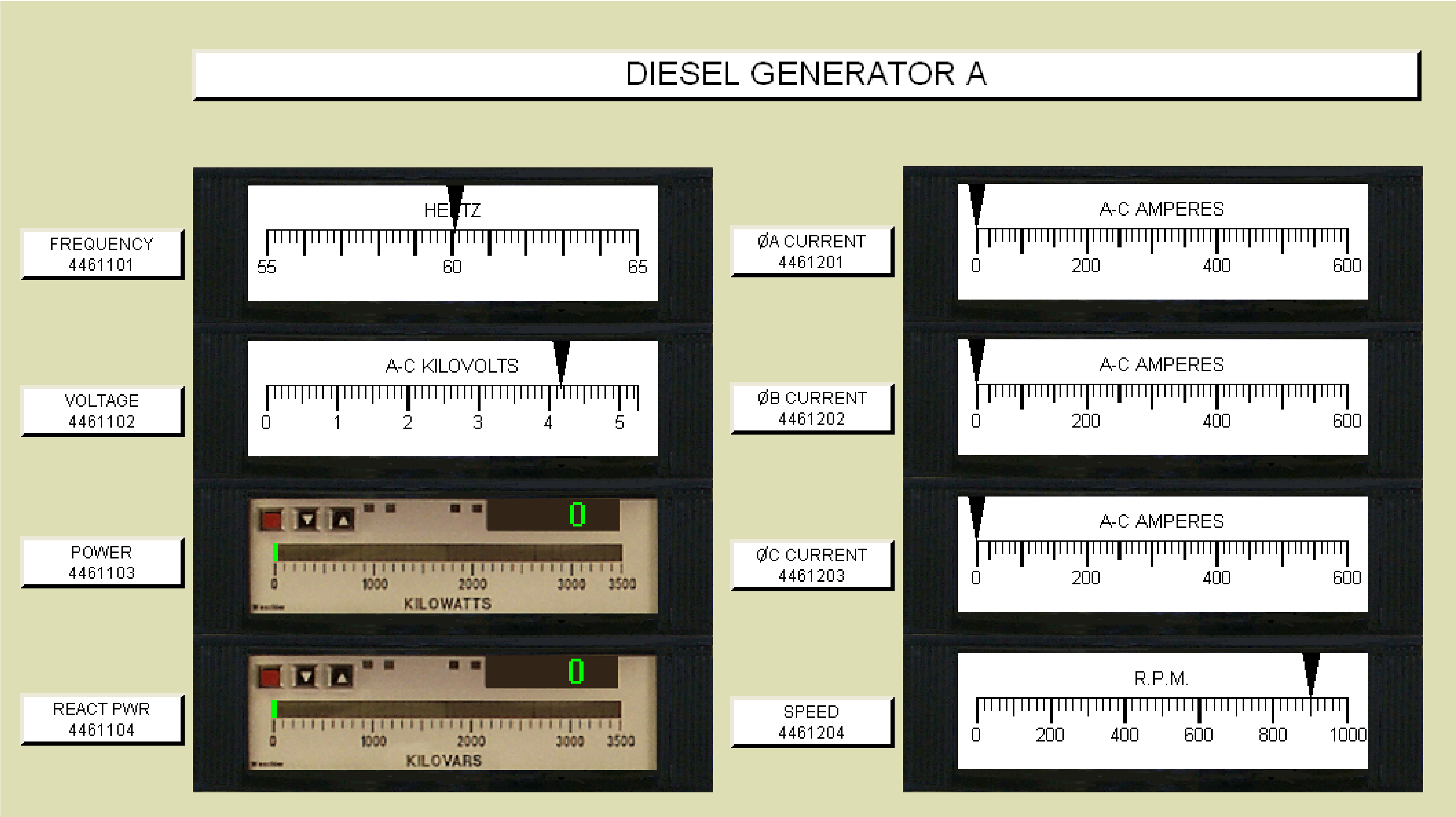
- Grid conditions (running)
- DG conditions (incoming)
- Synchroscope Running vs. Incoming.

A CORRECT — With these conditions, DG A will pick up load and the synchronizing lights will go out.

B INCORRECT — DG speed will not stabilize at a higher value.

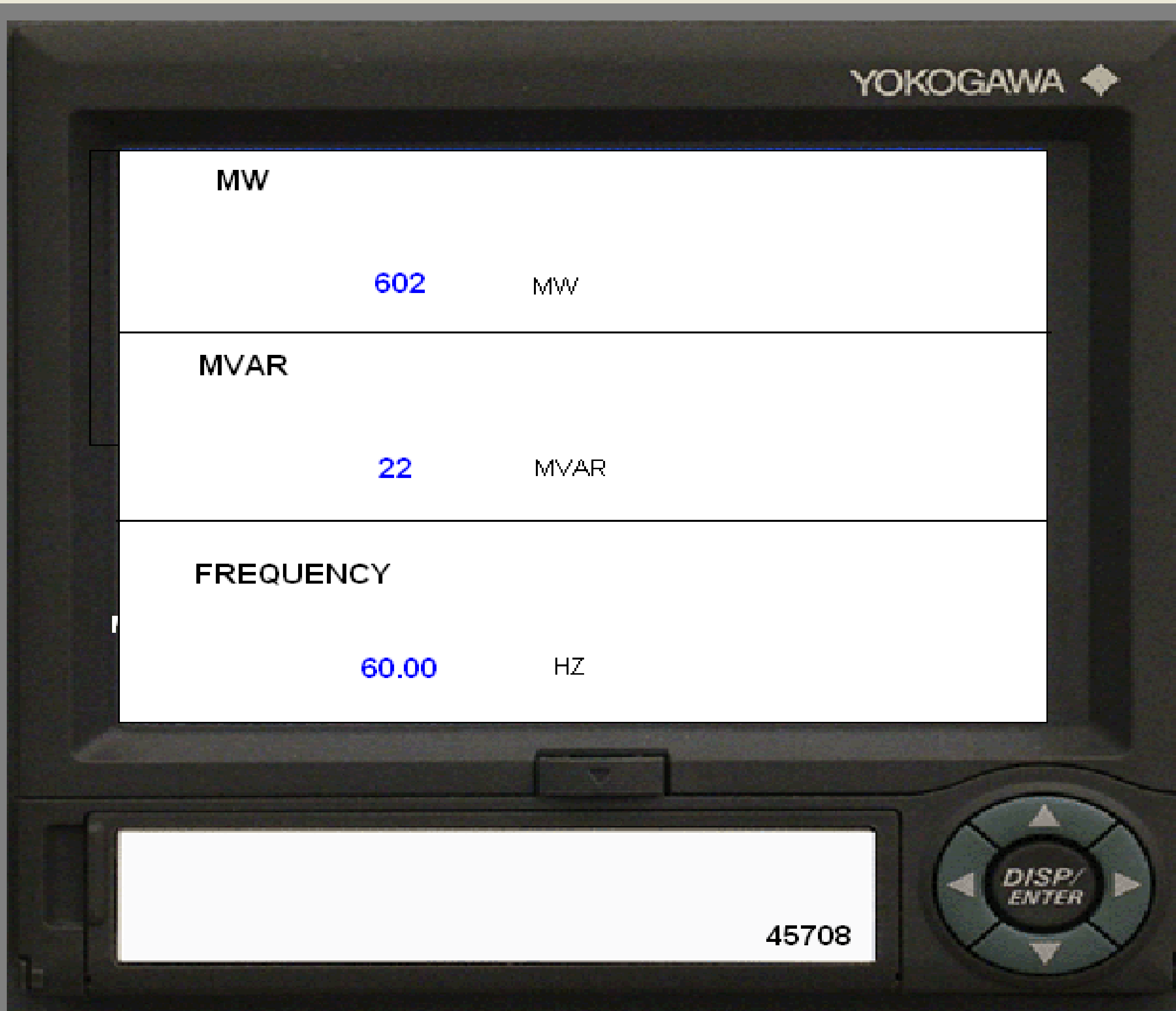
C INCORRECT — Breaker 1-509 will remain closed with the current conditions.

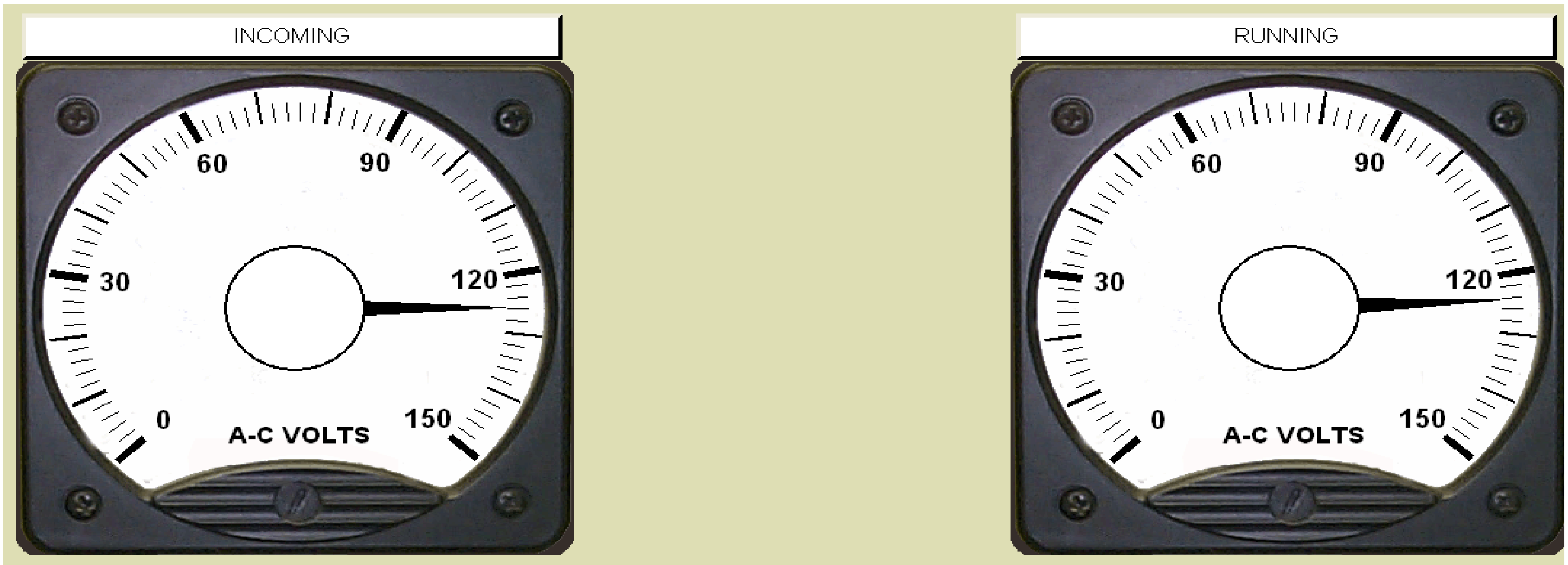
D INCORRECT — The output voltage may vary slightly but the synchronizing lights will be off after synchronization is complete.


DIESEL GENERATOR A
FREQUENCY
4461101
HERTZ
55
60
65
VOLTAGE
4461102
A-C KILOVOLTS
0
1
2
3
4
5
POWER
4461103
0
0
1000
2000
3000
3500
KILOWATTS
REACT PWR
4461104
0
0
1000
2000
3000
3500
KILOVARS
ØA CURRENT
4461201
A-C AMPERES
0
200
400
600
ØB CURRENT
4461202
A-C AMPERES
0
200
400
600
ØC CURRENT
4461203
A-C AMPERES
0
200
400
600
SPEED
4461204
R.P.M.
0
200
400
600
800
1000

MAIN GENERATOR

POWER




INCOMING
0
30
60
90
120
150
A-C VOLTS
RUNNING
0
30
60
90
120
150
A-C VOLTS

50. Given the following:

- N-RM-45, “Radiation Monitoring System”, is being performed to shutdown R-19, Steam Generator (SG) Blowdown Liquid Monitor.
- The Keyswitch for R-19 is positioned from ON to OFF.

Which of the following will result from this switch manipulation?

A. Annunciator 47015-B, RAD Monitor Power Supply Failure, actuates due to removing power to R-19.

B. Annunciator 47033-35, TLA-15 RMS Above Normal, actuates due to losing the communication link from R-19.

C. A High Radiation (fail) signal is generated, and the SG Blowdown and SG Sample Isolation Valves automatically close.

D. A High Radiation signal is defeated, preventing automatic closure of the SG Blowdown and SG Sample Isolation Valves.

Answer: C

Cognitive Level:
Memory 1-I: Recall of system response for operating process radiation monitor R-19.

K/A:
073A4.02 - Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel. RO/SRO IMP 3.7/3.7

Objective:
RO2-01-LP045.004: Describe the operation, automatic actions, interlocks, controls and alarms associated with the RADIATION MONITORING System and the following major system components: Process Radiation Monitors.

Reference:
N-RM-45, Radiation Monitoring System, step 4.3.17, R-19, S/G Blowdown Liquid
AOP-RM-001 Abnormal Radiation Monitoring System, Attachment A, Step A-10, S/G Blowdown Liquid Monitor Automatic Actions.
E-3748, Integrated Logic Diagram – Process-Radiation Monitors
ARP-47015-B, Rad Monitor Power Supply Failure
ARP-47033-35, TLA-15 RMS Above Normal
E-2010, Integrated Logic Diagram – Secondary Sampling System
E-2019, Integrated Logic Diagram – Radiation Monitoring
E-1629, Integrated Logic Diagram – Main Steam and Steam Dump System

Source:
Old Kewaunee Bank - Question 045ADV025
KNPP Initial RO2-01-LP045.004 #8.

Justification:

A INCORRECT The alarm is cause by a loss of power to PS-100, the power supply. Both power supplies remain functional when the keyswitch is positioned to OFF.

B INCORRECT TLA-15 does not actuate due to a communication loss. It is a plausible distracter because a high rad signal is generated and TLA-15 annunciates during high rad conditions.

C CORRECT Per N-RM-45 and AOP-RM-001, taking the keyswitch to off actuates automatic functions.

D INCORRECT Defeat automatic actuation is accomplished by taking the keyswitch to the KEYPAD position.

51. Given the following:

- The unit is at 8% NI power, performing a startup after a refueling outage.
- Preparations to begin rolling the Main Turbine are in progress.
- The Balance Of Plant operator identifies that P-486, Turbine Impulse Pressure Instrument, is reading off scale high.
- Shortly after the failure of P-486, RXCP A Trips.
- Annunciator 47012-J, RCS Loop A Flow Low, is LIT.

What is the expected Plant AND Operator Response?

	Plant Response	Operator Response
A.	Permissive P-7 will not cause a reactor to trip	The operators will manually trip the reactor
B.	Permissive P-8 will cause a reactor trip	The operators will perform the immediate actions for a reactor trip
C.	Permissive P-10 will not cause a reactor trip	The operators will perform a normal reactor shutdown
D.	Permissive P-13 will cause a reactor trip	The operators will manually trip the turbine

Answer: A

Cognitive Level:

Analysis 3-PEO: The operator must analyze the plant conditions to determine the plant response and required operator actions

K/A:

012A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the RPS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument power. RO/SRO IMP 3.6/3.9

Objective:

RO2-05-LP471.004 - Describe the operation, automatic actions, interlocks instrumentation, controls and alarms associated with the Reactor Protection System and the following major components

3. Reactor Trips

References:

AOP-RC-005, Abnormal RXCP Operation
ARP-47012-J, RCS Loop A Low Flow
AOP-MISC-001, Response to Instrument Failure
E-2051-1, Integrated Logic Diagram – Power Range Nuclear Instrumentation
XK-100-144, Logic Diagram – Reactor Trip Signals
XK-100-156, Logic Diagram – Turbine Trip Runbacks & other Signals
XK-100-147, Logic Diagram – Primary Coolant System Signals

Source:

New

Justification:

A CORRECT — P-7 would not cause a reactor to trip. The operators would have to manually trip the reactor, per 47012-J.

B INCORRECT — Permissive P-8 does not cause the reactor trip.

C INCORRECT — P-10 will not cause a reactor trip. Manual trip is required.

D INCORRECT — P-13 will not cause the reactor to trip.

52. Given the following:

- The plant is operating at 100% Rated Thermal Power.
- Turbine Building Service Water Header Selector switch is in the 1A position.
- SW Header ‘A’ Pressure, PI-41503, indicates 58 psig and stable.
- SW Header ‘B’ Pressure, PI-41506, indicates 75 psig and stable.
- Forebay level is 50% and stable.

What procedure entry requirements are met?

A. The entry conditions for AOP-GEN-001, “Operator Immediate Actions”, are met and the crew should perform the required immediate action.

B. The entry conditions for AOP-SW-001, “Abnormal Service Water System Operation are met and the crew should perform the procedure.

C. Annunciator 47051-Q, Turbine Building Service Water Isolation, should be LIT and the crew should perform the actions of the ARP.

D. Annunciator 47052-M, Forebay Level Low, should be LIT and the crew should perform the actions of the ARP.

Answer: B

Cognitive Level:

Analysis 3-SPK: Using the given plant information, determine if procedure entry requirements are met.

K/A:

076 G2.4.4 - Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. RO/SRO IMP 4.5 / 4.7

Objective:

RO4-03-LPD11.005 – In accordance with OP-KW-AOP-SW-001, Abnormal Service Water System Operation, SUMMARIZE the subsequent operator actions necessary to respond to the following:

a. Low Forebay Level
b. Low SW Header Pressure from SW Isolation
c. Low SW Header Pressure from Insufficient operating pumps
d. Low SW Header Pressure from System Leakage

Reference:

AOP-GEN-001, Immediate Operator Actions
ARP-47051-P, SW Header Pressure Low
ARP-47051-Q, Turbine Bldg Service Water Isolation
ARP-47052-M, Forebay Level Low
ARP-47052-P, Turbine Bldg SW Header Abnormal
ARP-47052-Q, Turbine Bldg SW Isolation Alert
AOP-SW-001, Abnormal Service Water System Operation

Source:

New

Justification:

A	INCORRECT	AOP-GEN-001, "Immediate Operator Actions", requires a reactor trip for CW Pump trip if the Turbine is latched, there are no CW Pumps running and forebay level is < 64%. The CW pumps never tripped, therefore is still running and entry into AOP-GEN-001 for Loss of CW Pumps would not be valid.
B	CORRECT	With the given conditions, Annunciator 47051-P, SW Header Pressure Low, 47052-P, Turbine Bldg SW Header Abnormal, and 47052-Q, Turbine Bldg SW Isolation Alert, will be in alarm. All Three are entry conditions for AOP-SW-001.
C	INCORRECT	Annunciator is caused by both SW-4A and SW-4B being closed. SW-4A/4B close when the associated SW header pressure is less than 82.5 psig and Safety Injection. Annunciator 47052-Q, Turbine Bldg SW Isolation Alert, alarms when either SW header pressure is less than 82.5 psig.
D	INCORRECT	Annunciator 47052-M will alarm at 47% forebay level not 50%.

53. Given the following:

- The unit is at 8% power by Nuclear Instruments following a refueling outage.
- IA-1303, IA to MS-1B Mn Stm Isol Valve 1B, was inadvertently CLOSED.
- A small air leak develops on one of the MSIV 'B' air accumulators.
- Annunciator 47063-K, MSIV AIR ACMTR Pressure Low, is LIT.

Which of the following is an expected response of plant?

A. MS-1B, S/G B MSIV, will fail to CLOSE on a subsequent MSIV isolation signal.

B. The alternate air accumulator will maintain MS-1B, S/G B MSIV, OPEN.

C. A reactor trip will occur when MS-1B, S/G B MSIV, air pressure is LESS THAN 60 psig.

D. A turbine trip will occur when MS-1B, S/G B MSIV comes 3º off the full OPEN position.

Answer: D

Cognitive Level:

Comprehension 2-RI: Recognize the interaction between Instrument Air and the MSIV accumulators. Determine the response of the plant at 8% reactor power and what signal generates the turbine trip.

K/A:

078K1.05: Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: MSIV air.

Objective:

RO2-02-LP06A.004: Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Main Steam and Steam Dump System and the following major system components: Main Steam Isolation Valves.

Reference:

NCL-MS-001, Main Steam and Steam Dump Prestartup Checklist
ARP-47063-K, MSIV Air ACMTR Pressure Low
OPERM- 213-7, Operation Flow Diagram – Station & Instrument Air System Sh 7

Source:

New

Justification:

A INCORRECT MSIVs fail closed on loss of Instrument Air.

B INCORRECT The alternate air accumulator will also depressurize due to the physical connection between the two air accumulators.

C INCORRCT A reactor trip will not occur below P-10 due to MSIV closure. Also the trip signal is not generated based on air pressure but based on MSIV position.

D CORRECT A loss of air pressure will cause the MSIV to fail close. At 3º off the open seat a turbine trip will be generated. Below P-10 a reactor trip will not be generated.

54. Given the following:

- The unit is operating at 100% Rated Thermal Power.
- Station and Instrument Air Compressor 'G' is OOS.
- Air Compressor 'C' is in OFF.
- Air Compressor 'F' is in service.
- Air Compressor 'F' trips on high HP air temperature.
- Instrument Air header pressure has lowered to 92 psig due to minor system leaks and is now stable.

Which of the following is the correct status for the equipment listed?

A. Air Compressor 'B' is running and loaded.
AND
SA-400, Station Air Header 1B Isolation, is partially closed.

B. Air Compressor 'B' is running and unloaded.
AND
SA-400, Station Air Header 1B Isolation, is fully open.

C. Air Compressor 'A’ is running and loaded.
AND
SA-400, Station Air Header 1B Isolation, is fully closed.

D. Air Compressor 'A is running and unloaded.
AND
SA-400, Station Air Header 1B Isolation, is modulating to maintain Instrument Air Header pressure > 92 psig.

Answer: A

Cognitive Level:

Memory 1-I: Recall/recognize the status of the Instrument Air system during lowering instrument air header pressure. Recall the setpoints for each air compressor for starting and loading. Recall the setpoints for station air header isolations SA-200 and SA-400.

K/A:

078A3.01 - Ability to monitor automatic operation of the IAS, including: Air pressure. RO/SRO Imp 3.1/3.2

Objective:

RO4-03-LPD17.001 - Describe the purpose, symptoms, immediate operator actions, and automatic actions associated with the following procedures: OP-KW-AOP-AS-001, Loss of Instrument Air.

Reference:

AOP-AS-001, Loss of Instrument Air
ARP-47051-I, Station & Instr Air System Fault

Source:

Bank- 2006 NRC Retake

Justification:

Per ARP-47051-I, Station and Instrument Air System Fault,

Step 3.4.4: If air header pressure decreases to 95 psig, then SA-200 and SA-400 start to close and will be fully closed at 90 psig.

Step 3.4.6: The following actions occur in response to lowering IA header pressure:

- At 105 psi, Air Compressors A and B start
- If in auto, Air compressor C will also start at 105 psi
- At 100 psi, Air Compressor C loads (100-110 psi)
- at 98 psi, Air compressor B loads (98-108 psi)
- At 96 psi Air Compressor A loads (96-106 psi)
- At 95 psi, SA-200, SA-400 starts to close
- At 90 psi, SA-200 and SA-400 are fully closed.

A CORRECT Compressor B starts at 105, loads at 98 psig. At 95 psig SA-400 begins to close, by 90 psig SA-400 is fully closed therefore SA-400 will be partially closed.

B INCORRECT Compressor B starts at 105, loads at 98 psig. At 95 psig SA-400 begins to close, by 90 psig SA-400 is fully closed therefore SA-400 will be partially closed.

C INCORRECT At 105 psi, Air Compressor A starts. At 96 psi Air Compressor A loads. At 95 psig SA-400 begins to close, by 90 psig SA-400 is fully closed therefore SA-400 will be partially closed.

D INCORRECT At 105 psi, Air Compressor A starts. At 96 psi Air Compressor A loads. At 95 psig SA-400 begins to close, by 90 psig SA-400 is fully closed therefore SA-400 will be partially closed.

55. Which containment personnel air lock malfunction affects both the containment LCO and the containment air locks LCO?

LCO 3.6.1 Containment shall be OPERABLE.

LCO 3.6.2 Two containment air locks shall be OPERABLE.

A. Total air lock leakage of 1.1 L_a.

B. Failure of the air lock interlock mechanism.

C. Obstruction prevents closing the outer air lock door.

D. Inability to lock closed both containment air lock doors.

Answer: A

Cognitive Level:

Memory 1-F: Know what components comprise the LCOs for containment and Air Locks.

K/A:

103K1.05 - Knowledge of the physical connections and/or cause-effect relationships between the Containment System and the following systems: Personnel access hatch and emergency access hatch. RO/SRO IMP 2.8/3.0

Objective:

RO2-04-LP056.006 - Identify the Containment System components with Technical Specifications.

References:

ITS, Section 3.6.1 and 3.6.2

ITS Bases, Section 3.6.1. and 3.6.2

Source:

New

Justification:

A CORRECT — When leakage from the containment air locks causes total containment leakage to be greater than 1.0 La then containment does not meet its LCO 3.6.1. For the airlock to be considered OPERABLE, the air lock must be in compliance with the Type B air lock leakage test making it able to meet LCO 3.6.2.

B INCORRECT — Failure of the air lock interlock mechanism does not cause a failure to meet LCO-3.6.1 for containment.

C INCORRECT — The inability to lock both air lock doors means that the airlocks do not meet LCO 3.6.2 for the air lock. Unless leakage exceeds 1.0 La then containment still meets LCO-3.6.1.

D INCORRECT — Failure of the ability to close one air lock door does not cause a failure to meet LCO-3.6.1 for containment.

56. Which of the following lists a criteria that the 400 ft^3 steam volume and 600 ft^3 water volume of the Pressurizer are designed to meet?

A. Keep the Pressurizer heaters covered during a 10% step load increase.

B. Prevent a reactor trip due to Pressurizer level with a 50% ramped load reduction and Steam Dumps NOT available.

C. During a reactor trip Pressurizer level will remain greater than 21% assuming steam dumps and/or Steam Generator relief valves lift as designed.

D. The pressure surge from a 30% ramped load reduction with only automatic Rod Control operation will not exceed the design capacity of both Pressurizer Safeties.

Answer: A

Cognitive Level:

Memory 1-F: Design fact about RCS for compensating expansion and contraction

K/A:

002K4.07 - Knowledge of RCS design feature(s) and/or interlock(s) which provide for the following: Contraction and expansion during heatup and cooldown. RO/SRO IMP 3.1/3.5

Objective:

RO2-01-LP36B.002 - Describe the PRZR and PRT systems. Include the following in the description:

2. Function/purpose, design basis, operating characteristics, and physical location as appropriate for the following major components:
    a. Vessel

Reference:

000-SD-36, System Description Reactor Coolant System
USAR 4.1.5.3 - 4.1.5.5

Source:

New

Justification:

A CORRECT	Step load increase of 10% will still keep the PRZR heaters covered.
B INCORRECT	The 50% load reduction is based upon steam dumps being available.
C INCORRECT	Criteria is the PRZR will not empty.
D INCORRECT	The design criteria is that a pressure surge from a 50% load reduction could cause the PRZR safeties to lift preventing a reactor trip.

57. Given the following:

- The reactor is initially critical below the Point of Adding Heat.
- Both Reactor Coolant Pumps are running.
- Main Steam Isolation Valves (MSIV) and MSIV bypass valves are closed.
- Moderator Temperature Coefficient is negative.
- RCS Tave is 547ºF and stable with heat removal through the Steam Generator (SG) PORVs
- Intermediate Range (IR) Nuclear Instruments N-35 and N-36 indicate 1x10-4% power.

If both SG PORVs fail CLOSED and NO operator action is taken, how will reactor power respond?

Reactor power will lower . . .

A. temporarily, then return to 1x10-4% power on Intermediate Range instruments due to subcritical multiplication.

B. to a subcritical power level and stabilize at a value in the source range based on subcritical multiplication.

C. initially, then return to 1x10-4% power on Intermediate Range instruments due to the resulting rise in RCS Tave.

D. until the positive reactivity from the temperature rise causes the reactor to stabilize at a critical power level in the Source Range.

Answer: B

Cognitive level:

Comprehension 2-RI: Determine based on the given conditions the resulting impact to reactor power and nuclear instruments.

K/A:

015K5.06 - Knowledge of the operational implications of the following concepts as they apply to the NIS: Subcritical multiplications and NIS indications. RO/SRO IMP 3.4/3.7

Objective:

PRO8.11 - Describe reactor power response once criticality is obtained.

Reference:

PWR Generic Fundamentals, Reactor Theory Chapter 8, Reactor Operational Physics

Source:

New

Justification:

A INCORRECT	With the PORV failed shut, the temperature of the RCS will rise, negative reactivity will be inserted due to the negative temperature coefficient, this will cause power will lower. This process makes the return to the original power level wrong.
B CORRECT	As temperature rises, negative reactivity is inserted because of the negative temperature coefficient until power is reduced to the source range where subcritical multiplication will take over.
C INCORRECT	Power will go down, not up with an increase in RCS temperature based on the negative temperature coefficient.
D INCORRECT	Since the Moderator Temperature is negative, positive reactivity cannot be inserted due to a temperature increase. Any temperature increase will result in the addition of negative reactivity. Subcritical multiplication will cause power to stabilize in the source range.

58. Given the following:

- The unit has been operating at 100% Rated Thermal Power for the last three months.
- The following annunciators are LIT:
    - 47033-12, TLA-2 RCS Subcooling High/Low
    - 47033-24, TLA-9 Core Exit T/C Tilt
    - 47044-F, ICCMS Panel Trouble

Which of the following plant conditions could cause ALL of the above alarms?

A. Train 'A' of Reactor Vessel Level Indication fails to 0%.

B. RCS Wide Range Pressure Channel PT-420 fails low.

C. One Core Exit Thermocouple is 671ºF and rising slowly.

D. A momentary reactor power excursion causes a spike to 102%.

Answer: C

Cognitive Level:

Memory 1-F: Recognize the conditions that would result in the given alarms.

K/A:

017K6.01: Knowledge of the effect of a loss or malfunction of the following ITM System Components: Sensors and detectors RO/SRO IMP 2.7/3.0

Objective:

RO2-05-LP050.004 - DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the In-core Instrumentation & Inadequate Core Cooling Monitor Systems and the following major system components: Core Exit Thermocouple Monitor

Reference:

ARP-47033-12, TLA-2 RCS Subcooling High/Low
ARP-47033-24, TLA-9 Core Exit T/C Tilt
ARP-47044-F, ICCMS Panel Trouble

Source:

New

Justification:

A	INCORRECT	A loss of train A Reactor Vessel level indication will cause an ICCMS Panel Trouble, TLA-9.
B	INCORRECT	RCS Wide Range will result in alarms, TLA-2 and ICCMS Panel Trouble, but not core exit T/C Tilt.
C	CORRECT	The failure of CETC to 671ºF will result in an indication less than 20ºF subcooling. It will also cause alarms for the CET Tilt and ICCMS Panel Trouble.
D	INCORRECT	The additional heat production will not cause less than 20ºF of subcooling or CET Tilt.

59. Which of the following correctly describes the combination of systems that are used to control Containment hydrogen concentration following a Design Basis LOCA?

A. Waste Gas system

AND

Containment Ventilation System

B. Post LOCA Train 'A'

AND

Containment Vacuum Breakers

C. Instrument Air System

AND

Shield Building Ventilation System

D. Containment Dome Vent Fans

AND

Auxiliary Building Ventilation System

Answer: C

Cognitive Level:
Memory 1-F: Knowledge of the systems used to control Containment hydrogen concentration following a DBA.

K/A:
028K1.01 - Hydrogen Recombiner and Purge Control System (HRPS)
Knowledge of the physical connections and/or cause-effect relationships between the HRPS and the following systems:
Containment annulus ventilation system (including pressure limits)
RO/SRO IMP 2.5/2.5

Objective:
RO2-04-LP18C.002 (3) - DESCRIBE the Reactor Building Ventilation Post LOCA H_2 Control System to include the following:

3. Interfaces with the following:
   a. Reactor Building Ventilation
   b. Instrument Air
   c. Hydrogen Gas Analyzer
   d. Auxiliary Building Ventilation
   e. Shield Building Ventilation

Reference:
000-SD-18, System Description Reactor Building Ventilation, Section 3.10
OPERM-403, Flow Diagram – Reactor Bldg Vent System Post LOCA Hydrogen Control
NOP-RBV-003, POST LOCA Hydrogen Control

Source:
ILT Master Bank RO2-04-LP18C.002
Old Kewaunee Bank - Question 0180000001K0201

Justification:

A. INCORRECT Waste Gas system is not used for post LOCA hydrogen control.

B. INCORRECT Vacuum breakers are for pressure control.

C. CORRECT The Instrument Air system is used to dilute the containment atmosphere and the Shield Building Ventilation system is used to vent Containment.

D. INCORRECT Containment Dome Fans are required to be in operation during venting of Containment for hydrogen and for hydrogen concentration monitoring. The Auxiliary Building Ventilation is only required for performing a normal Containment 2 inch vent.

60. Given the Following

- The unit is in MODE 6.
- Core Off-Load has NOT begun.
- Reactor Cavity water level = 23 ft above the top of the reactor vessel flange.
- Spent Fuel Pool (SFP) Pump ‘A’ is isolated for maintenance.
- ‘B’ Residual Heat Removal (RHR) Heat Exchanger is isolated due to a tube leak.
- A breaker fault caused a loss of power to the bus supplying power to SFP Pump ‘B’; the bus is NOT locked out.
- Annunciator 47055-N, Spent Fuel Pool Abnormal, is LIT.
    - SER 157, Spent Fuel Pool A Temperature High, is in ALARM.
    - SER 158, Spent Fuel Pool B Temperature High, is in ALARM.
- Local SFP Temperatures are 105ºF and slowly rising.

For the given indications, which of the following actions would you expect to be directed per AOP-SFP-001, “Abnormal Spent Fuel Pool Cooling and Cleanup System Operation”?

A. Close the crosstie Bkr from Bus 46 to provide power to SFP Pump ‘B’.

B. Align SFP Cooling to the RHR ‘A“ Heat Exchanger to provide cooling to the SFP.

C. Ensure Auxiliary Building Ventilation isolation is activated to minimize release to the environment.

D. Add makeup to the SFP to maintain SFP level between low level alarm and high level alarm to maintain boil off inventory.

Answer: D

Cognitive Level:

Analysis 3-SPK: The operator must hold the knowledge of MODE as well as required RHR alignments for that MODE. That knowledge must then be combined with the knowledge of the Spent Fuel Pool Cooling System.

K/A:

033A2.02 – Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SFPCS. RO/SRO IMP 2.7/3.0

Objective:

RO4-03-LPD04.001 – Describe the purpose, symptoms, immediate operator actions, and automatic actions associated with the following procedures: OP-KW-AOP-SFP-001, Abnormal Spent Fuel Cooling and Cleanup System

RO4-03SED04.005 – Respond to the following in accordance with OP-KW-AOP-SFP-001, Abnormal Spent Fuel Cooling and Cleanup System: Loss of Spent Fuel Pool Cooling

Reference:

AOP-SFP-001, Abnormal Spent Fuel Pool Cooling and Cleanup System Operation

ARP-47055-N, Spent Fuel Pool Abnormal

NOP-SFP-001, Spent Fuel Pool Cooling and Cleanup System, Section 5.3 and 5.5

E-259, Circuit Diagram – 480 v MCC 1-62D & 1-62E

ITS, Section 3.9.3

SDBD-KPS-ABV, System Design Basis Document for Auxiliary Building Ventilation System Section 2.1.3

Source:

New

Justification:

A INCORRECT With a loss of SFP cooling, neither SFP Pump running and power not available to SFP Pump 'A', the operator is directed to power bus 52 from bus 46 to provide power to SFP Pump 'A'. SFP Pump 'B' is powered from MCC 1-62E.

B INCORRECT This cannot be done, per NOP-SFP-001, alignment of the ‘A’ RHR Heat Exchanger to the SFP system will render it INOPERABLE, with the ‘B’ RHR Heat Exchanger INOPERABLE, and MODE 6, one RHR train is required. Note in TS 3.9.3 allows removal of the operating RHR system for $\leq$ 1 hour, This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the Refueling Cavity.

C INCORRECT This is done if R-13/14 are alarming which would be expected if the SFP level was low. Indications do not indicate low SFP Level. Also for high SFP temperatures direction is given to place the SFP Charcoal Filters in service.

D CORRECT Action in AOP-SFP-001 directs maintain the ability to cool the SFP by keeping the volume of water in the SFP for boil off.

61. Given the following:

- The unit has been at 100% Rated Thermal Power for 100 days.
- Fuel is being moved in the Spent Fuel Pool (SFP) North Pool in preparation to be loaded into a Dry Cask.
- The SFP canal is pumped down to support maintenance on the upender pivot pins.

If the South Pool Transfer Canal gate seal completely fails, and there are NO operator actions taken, what is the expected final level of the SFP?

SFP level will be approximately ____ feet below the Spent Fuel Pool floor.

A. 2

B. 4

C. 9

D. 17

Answer: C

Cognitive Level:

Knowledge 1-F: The operator must hold the knowledge of the pool level in relationship to an empty canal, the amount of water loss in the pool.

K/A:

034A1.02 – Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including: Water level in the refueling canal. RO/SRO IMP 2.9/3.7

Objective:

RO2-01-LP021.001 – Describe the function/Purpose, design basis, operating characteristics of the Spent Fuel Pool Cooling & Cleanup System

Reference:

AOP-SFP-001, Abnormal Spent Fuel Pool Cooling and Cleanup System Operation

Source:

New

Justification:

A INCORRECT This is the value out of the USAR for the pump suction.

B INCORRECT This is the approximate level of the bottom of the skimmer for the SFP.

C CORRECT Per the note in OP-KW-AOP-SFP-001 this is the correct answer.

D INCORRECT This level it too low, is possible if the student does not realize the containment transfer tube is isolated.

62. Given the following:

- The unit is at 100% Rated Thermal Power.
- Electrical systems are in a normal lineup when Main Transformer 'B' experiences a Sudden Pressure.
- Annunciator 47075-I, MAIN XFRM Sudden Pressure, is LIT.

Which of the following describes a plant effect of the above events?

A. Generator #1 Field Breaker trips causing Generator #1 Main Breaker G-1 to open

AND

A fast transfer of Buses 1-1, 1-2, 1-3, 1-4 and 1-5 to the Reserve Auxiliary Transformer

B. Generator #1 Main Breaker G-1 opens causing Diesel Generators 1A and 1B to start

AND

A reactor trip causes a subsequent turbine trip

C. Associated 63X Sudden Pressure Relays actuate and Diesel Generators 1A and 1B start

AND

Buses 1-5 and 1-6 transfer and load to their respective Diesel Generators

D. 86/T1A and 86/T1B Main Generator lockout relays actuate and Generator #1 Field Breaker trips

AND

A turbine trip signal is generated

Answer: D

Cognitive Level:

Comprehension 2-RI Recognize the plant response in multiple systems to a transformer fault and generator trip.

K/A:

045A3.11: Ability to monitor automatic operation of the MT/G system, including: Generator Trip. RO/SRO IMP 2.6/2.9

Objective:

RO2-03-LP043.004: DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls, and alarms associated with the Electrical Generation System and the following major components:

- Exciter
- Generator sync selector switch
- G-1, 43 switch
- G-1, 101 switch
- Field Supply breaker control switch
- Voltage regulator switch
- Base adjuster control switch
- Generator lockout relays
- Substation lockout relays
- Relays 86/T1C and 86/T1D

Reference:

ARP-47075-I, Main Xfmr Sudden Pressure
ARP-47072-I, Generator Lockout
E-910, Interlock Logic Diagram – Gen. Reserve Aux. Trans. & Ter. Aux. Trans.
DCR 1703 - Diesel Generators no longer start on Generator Lockout Relays.

Source:

Bank question: Modified the stem and all four answers.
KNPP Initial - RO2-03-LP043.004 002

Justification:

A INCORRECT Emergency Bus 1-5 does not fast transfer to the RAT on a Generator Lockout. Bus 1-5 will remain energized from the TAT.

B INCORRECT The Diesel Generators do not start. They previously did start on a Generator Lockout however the auto start was removed with DCR 1703.

C INCORRECT DGs no longer start on a Generator Lockout. The associated Sudden Pressure relays will actuate.

D CORRECT This is the correct answer per ARP 47072-I.

63. Given the following:

- A radioactive waste release of the Gas Decay Tank 'A' to the Auxiliary Building Vent is in progress when the following annunciators alarm:
    - TLA-15, RMS Above Normal
    - 47011-B, Radiation Indication High
    - 47012-B, Radiation Indication Alert
    - 47045-M, Waste Disposal Abnormal
- The HIGH alarm is LIT for the following Radiation Monitors
    - R-13, Aux Bldg Vent Exhaust Radiation Monitor, HIGH light is LIT
    - R-14, Aux Bldg Vent Exhaust Radiation Monitor, HIGH light is LIT.
- The NAO reports the release of Gas Decay Tank 'A' was automatically terminated.

What automatically terminated the release of Gas Decay Tank 'A' to the Auxiliary Building Ventilation System?

A. Waste Gas Compressor 'A' TRIPPED.

B. WG-36, Gaseous Waste Discharge Valve, CLOSED automatically.

C. WG-34A, Gas Decay Tank A Outlet Valve, CLOSED automatically.

D. The discharge is automatically realigned to the Dearated Drain Tank.

Answer: B

Cognitive Level:

Knowledge 1-P: Knowledge of the automatic actions associated with the Waste Gas system that activated by Radiation Monitors

K/A:

071A4.25 – Ability to manually operate and/or monitor in the control room: Setting of process radiation monitor alarms, automatic functions, and adjustment of setpoints. RO/SRO IMP 3.2/3.2

Objective:

RO2-01-LP045.004 – DESCRIBE the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the RADIATION MONITOR System and the following major system components:

- Process Radiation Monitors
- Area Radiation Monitors

Reference:

000-SD-45, System Description, Radiation Monitoring
000-SD-32B System Description, Gaseous Radioactive Waste Disposal System
E-2048, Integrated Logic Diagram – Waste Disposal System
AOP-RM-001, Abnormal Radiation Monitoring System
OPERXK-100-132, Flow Diagram – Waste Disposal System

Source:

Bank AOI-83-LP045 004-2 4
Exam AOI-09-EXA21R

Justification:

A INCORRECT Was Gas Compressor does not trip on a signal from R-13. The Waste Gas Compressor trips on moisture separator column level.

B CORRECT WG-36 will auto close on R-13 and/or R-14 alarm.

C INCORRECT WG-34A does not have an auto close signal.

D INCORRECT Realignment of the GDT discharge to the DDT is manual operation.

64. Which of the following maintenance activities would affect the status of LCO 3.7.8.?

LCO 3.7.8 - Two Service Water trains shall be OPERABLE.

A. Electrical Maintenance troubleshooting results in a loss of the indication of Service Water Pump amps.

B. I&C miscalibration results in the Circ. Water pump NOT tripping at the required Forebay Water level.

C. One train of Circ. Water Flooding Protection Trip Circuitry is rendered nonfunctional during an I&C Surveillance.

D. SW-1300B, Comp Cooling Heat Exchanger B Outlet, is identified closed during a maintenance tagout walkdown.

Answer: B

Cognitive Level:

Memory 1-F: Recall of the CW impact on SW OPERABILITY. Recognize the maintenance activities that impact the status of the SW LCO.

K/A:

075G2.2.36 - Circulating Water: Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operation. RO/SRO IMP 3.1/4.2

Objective:

RO2-02-LP004.006 - Identify the Circulating Water System components with Technical Specification Requirements.

References:

ITS Bases, Section 3.7.8, see the LCO section

Source:

New

Justification:

Per the Bases for LCO 3.7.8, LCO section: "In addition, the Forebay Water Level Trip System must be OPERABLE for the SW System to be considered OPERABLE."

A INCORRECT The SW Pump Amp indication would affect LCO 3.3.3. Post Accident Monitoring.

B CORRECT Per the Bases for 3.7.8, see above.

C INCORRECT Flooding Protection Trip Circuitry effects TRM 8.7.7, Flooding Protection - Circulating Water Pump Trip. There is no effect on SW. The trip on a TB flood protects TB equipment. The CW Pump trip on low forebay level ensures SW suction.

D INCORRECT SW-1300B is a normally closed valve from the outlet of the CC Heat Exchanger. SW-1306, controls SW flow through the CC Heat Exchanger.

65. Which of the following functions are required for the following LCO?

LCO 3.3.4 The Dedicated Shutdown System Functions shall be OPERABLE.

A. RCS Pressure Control
   Decay Heat Removal via the AFW system
   On-Site Power to support RCS inventory control

B. Technical Support Center Habitability
   Decay Heat Removal via the Steam Dumps
   Emergency DC power to support Instrumentation

C. Spent Fuel Pool Cooling
   Safety Injection via the Safety Injection Pumps.
   Service Water to support Emergency Diesel Generator operation

D. Long Term Reactivity Control
   Decay Heat Removal via the Steam Generator PORVs
   Component Cooling Water to support Containment Spray Pump operation

Answer: A

Cognitive Level:

Comprehension 2-DR: Recognizing the relationship between the Dedicated Shutdown System and redundant equipment available for placing and maintaining the unit in MODE 3. Comprehensive level because the operator has to distinguish the function of the required Safety Support System to discern the correct answer.

K/A:

086K3.01 - Knowledge of the effect that a loss or malfunction of the Fire Protection System will have on the following: Shutdown capability with redundant equipment. RO/SRO IMP 2.7/3.2

Objective:

ROI-01-LPTS1.005 - Given plant information, IDENTIFY the system components that are required by ITS Bases to be OPERABLE to meet an LCO

References:

ITS Bases, Section 3.3.4 Background & LCO
AOP-FP-002, Fire in an Alternate Zone

Source:

New

Justification:

A CORRECT — All items listed are in the bases for TS 3.3.4. Onsite Power is required for RCS inventory control: Charging Pumps.

B INCORRECT — Technical Support Center Habitability is not required. Plausible because the Shift Manager is directed to go to the TSC for a fire in an Alternate Zone per AOP-FP-002.

C INCORRECT — Safety Injection is not required per TS Bases 3.3.4. The Dedicated Shutdown System is required to place and maintain the unit in MODE 3. Spent Fuel Pool Cooling is not listed as a function in T.S bases 3.3.4.

D INCORRECT — Component Cooling is a required Safety support system for the functions listed in TS bases 3.3.4. Containment Spray function is not a listed in function in TS bases 3.3.4.

66. Given the following:

- The unit was operating at 100% Rated Thermal Power when the reactor tripped.
- The crew has completed the required steps of E-0, “Reactor Trip or Safety Injection” and transitioned to ES-0.1, “Reactor Trip Response
- Main Steam Isolation Valves are OPEN
- The Balance of Plant Operator reports:
  - MS-312A-1, Gland Seal Steam to MSR B1 & B2 Relief Vlvs is OPEN
  - MS-312B-1, Gland Seal Steam to MSR B1 & B2 Relief Vlvs is CLOSED

What actions will be required to mitigate this?

A. Condenser steam dump valves will be used for controlling RCS temperature.

B. ‘A’ Train Condenser Steam Dump valves are still available for RCS temperature control.

C. Main Steam Isolation Valves will be closed to reduce the excessive cooldown of RCS temperature.

D. Steam Generator PORVs or Atmospheric Steam Dump valves will be used to control RCS temperature.

Answer: D

OE:

This question is based on CAP033283 and CE017631, Isolation of Sealing Steam to MSR Relief Valve Complicated Reactor Trip Response 04/26/2006. The valve was isolated for maintenance and the work order was not properly prioritized, the plant tripped and the trip response was complicated due to a loss of vacuum.

Cognitive Level:

Application 3-PEO: Operator must assess the conditions and then decide and the actions to mitigate that condition.

K/A:

G2.1.39: Conduct of Operations. Knowledge of conservative decision making practices. RO/SRO IMP 3.6/4.3

Objective:

RO2-05-LP06C.002 – Describe the Steam Dump Control System to include the following in the description

- Function/purpose, design basis, and operating characteristics as appropriate for the following major components:
  - b. Turbine Trip Steam Dump Controller
  - c. Steam Pressure Steam Dump Controller
- Interfaces with the following
  - d. Condenser Pressure
  - e. Turbine Trip

RO2-05-LP06C.004 – Describe the operation, automatic actions, interlocks, instrumentation, controls and alarms associated with the Steam Dump Control System and the following major system components:

2. Turbine Trip Steam Dump Controller
3. Steam Pressure Steam Dump Controller

Reference:

OPERM-206, Flow Diagram – Bleed Steam and Heater Vents
E-1226, Integrated Logic Diagram – Main Steam and Steam Dump System
ES-0.1, Reactor Trip Response

Source:

New

Justification:

A INCORRECT Condenser Steam Dumps cannot be used because of Loss of Condenser Vacuum, they will be unavailable.

B INCORRECT The lowering of Condenser vacuum will affect both Condensers. Since there is no other valves lineup information given it is safe to assume that both Condensers are affected and neither Condenser Steam Dumps may be used.

C INCORRECT The failure will cause a Loss of Condenser Vacuum and not raise the cooldown rate.

D CORRECT The fact that MS-312B-1, Gland Seal Steam to MSR B1 & B2 Relief Vlvs is CLOSED while MS-312A-1, Gland Seal Steam to MSR B1 & B2 Relief Vlvs is OPEN, should alert the operator that one of the valves is malfunctioning. The purpose of the valves is to allow low pressure sealing steam to the MSR Relief Valves to prevent air in-leakage to the secondary system, when the pressures are low in the MSR header. With the lowering or loss of vacuum the impact is to the control of RCS temperature. This is the correct method of temperature control per ES-0.1

67. Which of the following lists the correct application of Continuous Action Steps (CAS) during the performance of Emergency Operating Procedures?

A. Does NOT apply after a transition to another Emergency Operating Procedure.

B. Applies until the action stated in the step has been completed after which its applicability ends.

C. Applies to the entire procedure and its applicability continues when transitioning to a Normal Operating Procedure.

D. Applies to the current procedure only after the step is first procedurally encountered, and may apply after a transition is made to another Emergency Operating Procedure.

Answer: D

Cognitive Level:
Memory 1-P: Knowledge of the application of procedure notes, cautions, and continuous action steps to emergency operating procedure usage.
K/A:
G2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation. RO/SRO IMP 4.3/4.4
Objective:
RO4-04-LP001.007 - Explain the requirements of EOP implementation in accordance with OP-AP-104.
Reference:
OP-AP-104, Emergency and Abnormal Operating Procedures, section 3.4.3
Source:
New

Justification:

A INCORRECT (CAS) may apply after transition.

B INCORRECT (CAS) applies even after an action in the step has been completed.

C INCORRECT (CAS) are only applicable during transitions to other EOPs and are only applicable after the step is encountered in the procedure.

D CORRECT This statement is correct.

68. Which of the following lists components that can ALL be operated from the Dedicated Shutdown Panel?

A. Fire Pump 'A'
Fuel Oil Transfer Pump 'A'
CVC-15, CHRG Line To PRZR Aux Spray

B. Charging Pump 'A'
Battery Room Fan Coil Unit 'A'
LD-13A, Reactor Coolant Filter Bypass Valve

C. RHR pump 'A'
BT-3A, STM GEN 1A Blowdown Isol
LD-4B, Regen HX LTDN OTLT ORIF 1B Isol

D. Containment Fan Coil Unit 'B'
Service Water Pump 1B1
RHR-11, RHR Return to Loop B Cold Leg Isol

Answer: A

Cognitive Level:

Memory 1-S: Structures and locations: The student must identify the components on the DSP from memory

K/A:

G2.1.30 - Conduct of Operations: Ability to locate and operate components, including local controls. RO/SRO IMP 4.4./4.0

Objective:

RO2-05-LP034.004 - Describe the operation, automatic action, interlocks, instrumentation, CONTROLS, and alarms associated with the Residual Heat Removal System and the following major components:

1. Loop A(B) hot leg suction valves, RHR-1A(B), 2A(B)
5. RHR pumps

RO2-02-LP008.004 - Describe the operation, automatic action, interlocks, instrumentation, CONTROLS, and alarms associated with the Fire Protection System and the following major components:
Fire Pumps A(B)

RO2-03-LP42A.004 - Describe the operation, automatic action, interlocks, instrumentation, CONTROLS, and alarms associated with the Fire Protection System and the following major components:

RO2-05-LP035.004 - Describe the operation, automatic action, interlocks, instrumentation, CONTROLS, and alarms associated with the Chemical and Volume Control System.

RO2-04-LP016.004 - Describe the operation, automatic action, interlocks, instrumentation, CONTROLS, and alarms associated with the Turbine Building and Screen House Ventilation System and the following major components:

8. Dedicated Shutdown Panel (DSP)
    a. Local/Remote Switches (Three) and Indicating Lights
        1) Battery Room Fan Coil 1A

RO2-02-LP07A.004 - Describe the operation, automatic action, interlocks, instrumentation, CONTROLS, and alarms associated with the Steam Generator Blowdown and Blowdown Treatment System and the following major components:

1. Containment Isolation Valves BT-2A, 2B, 3A, and 3B

RO2-04-LP018.004 - Describe the operation, automatic action, interlocks, instrumentation, CONTROLS, and alarms associated with the Reactor Building Ventilation System and the following major components:

1. Containment Fan Coil Unit A (B) C (D)

References:

E-1619 Integrated Logic Diagram – Fire Protection System: Fire Pumps
E-1622 Integrated Logic Diagram – Diesel Generator Mech System:
E-2016 Integrated Logic Diagram – Turb Bldg & Scrn Hse Vent Sys:
E-3104 Integrated Logic Diagram – Reactor Building Vent System:
E-3000 Integrated Logic Diagram – Chemical Volume Control System:
E-1629 Integrated Logic Diagram – Main Steam & Steam Dump System:
E-2025 Integrated Logic Diagram – Chemical Volume Control System:
E-2036 Integrated Logic Diagram – Residual Heat Removal System:
E-2023 Integrated Logic Diagram – Chemical Volume Control System:

Source:

New

Justification:

A CORRECT All components can be operated from the DSP.

B INCORRECT Charging pump A cannot be operated from the DSP. Plausible because it is powered from the same source as other 'A' train components but is not designated as an 'A' train component for the DSP.

C INCORRECT BT-3A is a 'B' train powered component. It is plausible because is a redundant isolation for the 'A' S/G blowdown and 'A' S/G is used for plant cooldown from the DSP.

D INCORRECT SW pump 1B1 is a B train component. Service water pumps 1A1 and 1A2 are the 'A' train components.

69. Given the following:

- The unit is operating at 100% Rated Thermal Power when both SI Accumulators are declared INOPERABLE per LCO 3.5.1
- The Unit Supervisor enters LCO 3.0.3.

LCO 3.5.1 Two SI accumulators shall be OPERABLE

Which of the following combinations of Reactor Coolant System Temperature, and Pressure allows exiting LCO 3.0.3. in the SHORTEST amount of Time?

(Assume that the SI Accumulators remain INOPERABLE)

	RCS Temperature	RCS Pressure	Time Since INOPERABILITY
A.	199ºF	300 psig	34 hours
B.	350ºF	400 psig	12 hours
C.	450ºF	600 psig	6 hours
D.	540ºF	2235 psig	5 hours

Answer: C

Cognitive Level:

Comprehension 2RI: Relate the following pieces of information: Applicability of TS 3.5.1, Temp versus MODE, and Time requirements for LCO 3.0.3. Use that information to determine the correct choice of minimum time in choices listed to met the requirements of LCO 3.0.3

K/A:

2.2.40 - Ability to apply Technical Specifications for a system. RO/SRO IMP 3.4/4.7

Objective:

ROI-01-LPTS2.005 - Given plant conditions, APPLY the rules of ITS section 3.0 to ensure compliance with Improved Technical Specifications.

Reference:

ITS, LCO 3.0.2., LCO 3.0.3., and Section, 3.5.1.

Source:

New

Justification:

A INCORRECT — Applicability for TS 3.5.1 is MODE 3 > 1000 psig. The crew does not have to go to 199ºF which is MODE 5, only to MODE 3 < 1000 psig.

B INCORRECT — MODE 4 is not required. See A.

C CORRECT — In MODE 3 less than 1000 the T.S. 3.5.1 is no longer applicable therefore there is no longer a need to be in LCO 3.0.3. due to the rules of usage of LCO 3.0.2.

D INCORRECT — Since the plant is currently in MODE 3 with pressure greater than 1000 psig, TS 3.5.1. applicability is still met, therefore the requirements of LCO 3.0.3. are still applicable.

70. Which of the following is an example of acceptable preconditioning before performing a Technical Specification surveillance?

A. Removing electrical loads prior to a load rejection test.

B. Lubricating valve stems prior to as found stroke time testing.

C. Performing operational readiness checks prior to performance of the surveillance.

D. Performance of maintenance activities prior to a surveillance test with the intent of ensuring favorable test results.

Answer: C

Cognitive Level:
Memory 1-P: Knowledge of preconditioning.
K/A:
2.2.17 - Equipment Control/Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator. RO/SRO IMP 2.6/3.8
Objective:
RO4-01LPA10.003 Discuss the control room operating personnel responsibilities associated with Plant Control and Work Management for the following: WM-KW-100-1001, Work Management
Reference:
WM-KW-100-1001, Work Management
Source:
New

Justification:

A INCORRECT Example of unacceptable preconditioning from section of 5.2.4 of WM-KW-100-1001.

B INCORRECT Example of unacceptable preconditioning from section of 5.2.4 of WM-KW-100-1001.

C CORRECT Example of acceptable preconditioning from section of 5.2.4 of WM-KW-100-1001.

D INCORRECT Example of unacceptable preconditioning from section of 5.2.4 of WM-100-1001.

71. Why is it preferred that Post Accident Leakage Control System be activated per AOP-MDS-002, "Post Accident Leakage Control System", prior to establishing containment sump recirculation during a Large Break LOCA?

A. To prevent boron dilution of Containment Sump 'B' during containment sump recirculation.

B. The Post Accident Leakage Control System is required to obtain containment sump samples during containment sump recirculation.

C. High radiation levels could prohibit diverting the Deaerator Drain Tank Vent to containment during containment sump recirculation.

D. A large differential pressure across CVC-215B, Seal Water Filter Bypass valve, will develop during containment sump recirculation and prevent it from opening.

Answer: C

Cognitive Level

Memory 1-B: Knowledge of purpose of Post Accident Leakage Control System.

K/A:

G2.3.14 - Knowledge of the radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities. RO Imp / SRO Imp - 3.4 / 3.8

Objective:

RO4-04-LP018.002 - Summarize the purposes or bases of the following items as they relate to E-1, Loss of Reactor or Secondary Coolant: All Procedure Steps.

Reference:

OP-KW-AOP-MDS-002, Post Accident Leakage Control, NOTE Step 2
E-1, Loss of Reactor or Secondary Coolant, Note Step 19
BKG-E-1, Background Loss of Reactor or Secondary Coolant, KPS Step 19

Source:

BANK: 2009 ILT Written Exam #72

Justification:

A INCORRECT Dilution of the sump is not a concern at this time.

B INCORRECT Though the sample lines are the source of the radiation, the system does not need to be activated to sample from the High Rad Sample Room.

C CORRECT During a LB LOCA radiation from back leakage of sample lines may prohibit access to WG-309, DDT vent to containment.

D INCORRECT Containment Sump Recric does not cause a D/P across CVC-215B, though the valve is operated in AOP-MDS-002. CVC-215B is opened to lower Aux Building rad levels by keeping particles out of the filters.

72. An area in the Auxiliary Building has the following radiation and contamination values:

- Alpha particle Surface Contamination Level of 80 dpm/100cm^2
- General Area Radiation Level of 1200 mRem/Hr
- Airborne Radioactivity Level of 7 DAC - hours

Which of the following area postings should be displayed at the entrance to this area?

A. High Radiation

<u>AND</u>

Contaminated Area

B. Very High Radiation

<u>AND</u>

Airborne Radioactivity Area

C. Locked High Radiation

<u>AND</u>

Contaminated Area

D. Very High Radiation

<u>AND</u>

Contaminated Area

<u>AND</u>

Airborne Radioactivity Area

Answer: C

Cognitive Level:
Knowledge 1-D: The operator must hold the knowledge of the posting requirement.
K/A:
G2.3.4 – Radiological Controls: Knowledge of radiation exposure limits under normal or emergency conditions RO/SRO IMP 3.2/3.7
Objective:
RWT-09: Identify the posting associated with radiological areas.
Reference:
RP-AA-202, Radiological Postings
Source:
Bank, Master LXR Test Bank

Justification:

A INCORRECT — High Radiation Area has radiation levels in excess of 100 mrem/hr at 30 cm. Contamination Area has removable contamination greater than or equal to 1000 dpm/100cm^2 beta gamma, or 20 dpm/100cm^2 alpha.

B INCORRECT — Very High Radiation Area has radiation levels in excess of 500 rads in one hour at 1 meter. Airborne Radioactivity Area has concentration in excess of 2000 DAC.

C CORRECT — Locked High Radiation Area has radiation levels in excess of 1000 mrem/hr at 30 cm. Contamination Area has removable contamination greater than or equal to 1000 dpm/100cm^2 beta gamma, or 20 dpm/100cm^2 alpha.

D INCORRECT — Very High Radiation Area has radiation levels in excess of 500 rads in one hour at 1 meter. Contamination Area has removable contamination greater than or equal to 1000 dpm/100cm^2 beta gamma, or 20 dpm/100cm^2 alpha. Airborne Radioactivity Area has concentration in excess of 2000 DAC.

73. For which of the following tasks is a General RWP allowed?

A. To control work in areas having dose rates > 200 mrem/hour at 30 cm.

B. Entry into a Locked High Radiation Area or Very High Radiation areas for the sole purpose of performing a plant inspection.

C. Plant Inspections in High Radiation Areas when the radiological conditions are static and approved by the Manager of Radiological Protection and Chemistry.

D. Plant inspections where the expected dose exceeds 100 mrem per individual per entry when approved by the Manager of Radiological Protection and Chemistry.

Answer: C

Cognitive Level:
Memory1-P: Knowledge of RWPs.
K/A:
G2.3.12 – Radiological Controls: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc. RO/SRO IMP 3.2/3.7
Objective:

Reference:
RP-AA-106, Section 3.2 RWP Types and Scope.
Source:
New

Justification:

A INCORRECT Listed as a requirement for Specific RWP.

B INCORRECT Listed as a requirement for Specific RWP.

C CORRECT Per 3.2.4 a General RWP is allowed for this circumstance.

D INCORRECT Listed as a requirement for Specific RWP.

74. You are a licensed Reactor Operator working on outage tagouts in the Work Control Center; this is your relief week. The following plant announcement is made:

“Attention all personnel, Attention all personnel. We are experiencing an ALERT. Emergency Response Organization personnel should report to their duty locations. All other personnel should report to the nearest assembly area.”

To which of the following locations do you report?

A. The Control Room

B. Site Relocation Facility

C. Admin Assembly Room

D. Warehouse Annex Lunch Room

Answer: A

Cognitive Level:
Knowledge 1-P: The operator must possess the knowledge of where to report given the initial conditions.
K/A:
G2.4.39 – Emergency Procedures/Plan: Knowledge of RO responsibilities in emergency plan implementation. RO/SRO IMP 3.9/3.8
Objective:
RO4-02-LPD12.004 – Describe the expected response for off-shift Control Room staff when notified of a plant emergency.
Reference:
EPIP-AD-01, personnel Response to the Plant Emergency, section 5.3.2
Source:
New

Justification:

The control room is the correct location for licensed reactor operators.

75. Given the following:

- The unit was operating at 100% Rated Thermal Power when a fault caused a failure of the Reserve Auxiliary Transformer.
- The following annunciators are LIT after the automatic response of the plant; NO operator actions have been taken.
  - 47013-A, Rad Monitor Sampling Flow High/Low
  - 47041-J, Charging Pump Trip
  - TLA-11, Reactor Thermal Power High
  - 47062-M, AFW Pump B Abnormal
  - 47082-E, RAT Lockout
  - 47082-G, Rat Differential Current
  - 47084-A, Substation Major

Based on the given annunciators in alarm, which of the following actions has priority?

A. Stop AFW Pump 'B'

B. Establish charging flow

C. VPL down to lower power

D. Align R-21 to sample containment

Answer: C

Cognitive Level:

Analysis 3-SPK: Using knowledge of plant response and given indications analyze the conditions to determine which action has priority.

K/A:

G2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm. RO/SRO IMP 4.1/4.3

Objective:

RO4-03-LPD14.003 - In accordance with the appropriate alarm response procedures, Summarize the required operator actions that are necessary to respond to the following: RAT Lockout.

Reference:

ARP-47013-A, Rad Monitor Sampling Flow High/Low
ARP-47041-J, Charging Pump Trip
ARP-47033-31, TLA-11, Reactor Thermal Power High
ARP-47062-M, AFW Pump B Abnormal
ARP-47082-E, RAT Lockout
ARP-47082-G, Rat Differential Current
ARP-47084-A, Substation Major

Source:

New:

Justification:

Give the initial list of annunicators, the RAT locked out, which means that bus 6 lost power until the ‘B’ Diesel Generator started supplying power to the bus. This transient will cause an uncoupling of the Heater Drain Pump, this will cause a low pressure to the Feedwater Pump suctions thus causing C-13 to open.

A INCORRECT AFW Pump 'B' will trip on low discharge pressure when it automatically starts with the blackout sequence. The tripping of AFW pump 'B" is the cause of the annunciator 47062-M, AFW Pump B Abnormal.

B INCORRECT Charging Pump 'B' will trip will loss of the RAT. Either Charging Pump 'A' or 'C' will still be running, both are ‘A’ train pumps.

C CORRECT C-13 will open during the transient, bypassing feedwater heaters. This will cause Thermal Power to rise. The operator will have to lower power.

D INCORRECT R-21 is already aligned to sample containment. The operators would be directed to re-start the sample pump for R-11/12.