1. 001 002/RO/OK/DIRECT/NRC EXAM 2004-033-2/00WE12 EK1.2/3.5/3.8/H/3

Operators are performing 2-OHP-4023-ECA-2.1, Uncontrolled Depressurization of All Steam Generators due to a steam leak inside containment along with failure of all SG stop valves to close.

Given the following plant conditions:

- Cooldown rate is stable at 155° F per hour.
- RCS cold leg temperatures are 340° F and lowering.
- Containment pressure is 8 psig and lowering.
- Narrow range Steam Generator levels indicate offscale low.
- Steam Generator AFW flow indicates $100x10^3$ pph to each SG.

Which ONE of the following choices is correct for these plant conditions?

A. Adjust AFW flow to $60x10^3$ pph on each Steam Generator.

B. Adjust AFW flow to $25x10^3$ pph on each Steam Generator, AFTER at least one SG narrow range level is greater than 13%.

C.✓ Adjust AFW flow to $25x10^3$ pph on each Steam Generator.

D. Isolate AFW flow to ONLY three of the Steam Generators.

ANSWER: C

A. INCORRECT. The $240x10^3$ pph ($60x10^3$/SG) is the normal minimum required for heat sink. With the reduced RCS temperature and cooldown rate this is not required at this time.

B. INCORRECT. Flow is throttled regardless of level. The minimum is $25x10^3$ pph when < 13% (Note the number is 28% for Adverse Containment which applies in this case.).

C. CORRECT. AFW flow is throttled to $25x10^3$ pph on each Steam Generator if the cooldown rate is > 100°F per hour and level is less than 28% (Adverse Containment).

D. INCORRECT. A minimum is $25x10^3$ pph is required to each SG when < 13% to minimize thermal shock.(Note the number is 28% for Adverse Containment which applies in this case).

LESSON PLAN/OBJ:RO-C-EOP07/#8
REFERENCE: 02-OHP-4023-ECA-2.1, Uncontrolled Depressurization of All Steam Generators Step 2 pg. 4

KA - 00WE12 EK1.2
Uncontrolled Depressurization of all Steam Generators
Knowledge of the operational implications of the following concepts as they apply to the Uncontrolled Depressurization of all Steam Generators:
Normal, abnormal and emergency operating procedures associated with Uncontrolled Depressurization of all Steam Generators
RO - 3.5 SRO - 3.8
CFR - 41.8 / 41.10 / 45.3
SCLR - 3SPK.

KA Justification - Requires the knowledge of the operational implications and required AFW flow rate actions in the EOPS associated with an Uncontrolled Depressurization of all SGS.

Question Source - Cook NRC Exam 2004-033-2, 21543-KEWAUNEE02

2. 002 025/RO/OK/MODIFIED/DIABLO2002-22392/000027 AK1.01/3.1/3.4/H/3

Given the following plant conditions on Unit 2:

- Reactor is at 100% power.
- All loop Tcolds are 540°F.
- All loop Thots are 606°F.
- Average of the 5 highest Core exit TCs is 620°F.
- A malfunction occurred with the pressurizer pressure control system resulting in an RCS pressure of 1915 psig without a change in reactor power.

Which ONE of the following describes the change from normal full power values to RCS subcooling based on the pressure transient?

Subcooling will lower to:

A. ✓ 11°F

B. 22°F

C. 58°F

D. 91°F

ANSWER: A

A. CORRECT. Saturation Temperature for 1930 psia (1915 psig) is ~631°F. Comparing thermocouples to Tsat yeilds. 631°F - 620.00°F = 11°F.

B. INCORRECT. Incorrectly uses Thot rather than T/C for calculation and subtracts 15 psi instead of adding.

C. INCORRECT. Incorrectly uses Tave (573°F) rather than T/Cs for calculation.

D. INCORRECT. Incorrectly uses Tcold rather than T/Cs for calculation.

Pressurizer Vapor space temperature is 650 at 2235 psig.

LESSON PLAN/OBJ: RO-C-GF13/#8
REFERENCE: RO-C-GF13

KA - 000027 AK1.01
Pressurizer Pressure Control (PZR PCS) Malfunction
Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions:
Definition of saturation temperature
RO - 3.1 SRO - 3.4
CFR - 41.8 / 41.10 / 45.3
SCLR - 3SPR

KA Justification - Requires the knowledge of the definition of saturation temperature (& subcooling) and the ability to relate the definition to a malfunction of the Pressuizer Pressure Control System resulting in a reduction in pressure.

Question Source - INPO Bank #22392 - DIABLOC-1012002
Original Question KA - ..027.AK1.01

Changed values values in Stem & Distractors.

3. 003 002/RO/OK/DIRECT/NRC EXAM 2004-016-2/00WE04 EK1.3/3.5/3.9/F/3

During a Large Break LOCA, an evaluation of plant status is made during Step 11 of 1-OHP-4023-E-1, Loss of Reactor or Secondary Coolant. Part of this evaluation includes a check of ECCS pump compartment sump alarms and auxiliary building vent stack and area radiation monitors.

Which ONE of the following reasons describes the BASIS for checking these alarms and radiation monitors in this procedure?

A. Determine if local actions can be performed without excessive personnel exposure.

B. Determine if ECCS leakage exceeds that assumed in the Control Room dose analysis.

C.✓ Determine if a transition should be made to address a LOCA outside of Containment.

D. Collect current radiation values to assist in Emergency Event classification.

ANSWER: C

A. INCORRECT. In-Plant operators are dispatched with radiation protection techs that assess the plant conditions with hand held instruments.
B. INCORRECT. Ongoing plant leakage from ECCS equipment is tracked to ensure that assumptions are met.
C. CORRECT. Plant sump alarms and radiation monitors are both checked to identify leakage in the auxiliary building. This check is made to determine if the operator should make a transition to 1-OHP-4023-ECA-1.2, LOCA Outside Containment.
D. INCORRECT. This assessment is done outside of the emergency operating procedure set (EOPs).

LESSON PLAN/OBJ: RO-C-EOP09/36
REFERENCE: PSBD Rev. 2, 12-OHP-4023-E-1 Background Document, EOP Step 11 Basis pg. 27

KA - 00WE04 EK1.3
LOCA Outside Containment
Knowledge of the operational implications of the following concepts as they apply to the LOCA Outside Containment:
Annunciators and conditions indicating signals, and remedial actions associated with the LOCA Outside Containment
RO - 3.5 SRO - 3.9
CFR - 41.8 / 41.10 / 45.3
SCLR - 1B

KA Justification - The question tests the knowledge of the basis behind checking the sump alarms and vent stack radiation monitors. The transition to another procedure (LOCA Outside Containment) to address the problem is the operational implication.

Question Source - NRC EXAM 2004-016-2

4. 004 004/RO/OK/DIRECT/ROBINSON2004-28135/00WE05 EK2.1/3.7/3.9/H/3

Given the following plant conditions:

- A Reactor Trip and Safety Injection have occurred.
- While implementing 2-OHP-4023-E-0, Reactor Trip or Safety Injection Step 10, it is determined that AFW flow can NOT be established.
- All SG NR levels are off-scale low.
- All SG WR levels are 58% and lowering.
- The crew has just entered 2-OHP-4023-FR-H.1, Response to Loss of Secondary Heat Sink.
- RCS Pressure is 175 psig and stable.
- Intact SG pressures are 475 psig and trending down.

Which ONE of the following describes the plant conditions and action required?

Steam Generators are:

A. required to provide secondary heat sink. Initiate Bleed and Feed per 2-OHP-4023-FR-H.1.

B. NOT required to provide secondary heat sink. Go to 2-OHP-4023-ES-0.0 Rediagnosis.

C. required to provide secondary heat sink. Remain in 2-OHP-4023-FR-H.1 to establish AFW Flow.

D.✓ NOT required to provide secondary heat sink. Return to 2-OHP-4023-E-0.

ANSWER: D

A. INCORRECT. Secondary heat sink is not required if SGs are at a higher pressure than the RCS. They act as a heat source.
B. INCORRECT. If SGs are NOT required for heat sink, the crew will return to E-0.
C. INCORRECT. SGs are NOT required, because RCS pressure is below SG pressure
D. CORRECT. Secondary heat sink is not required if SGs are at a higher pressure than the RCS. They act as a heat source. Crew returns to E-0.

LESSON PLAN/OBJ: RO-C-EOP11/#10
REFERENCE: 2-OHP-4023-E-0, 2-OHP-4023-FR-H.1, 12-4023-FR-H.1, Step 1

KA - 00WE05 EK2.1
Loss of Secondary Heat Sink
Knowledge of the interrelations between the Loss of Secondary Heat Sink and the following:
Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features
RO - 3.7 SRO - 3.9
CFR - 41.7 / 45.7
SCLR - 2RI

KA Justification - Requires the knowledge of the interrelationship between a loss of heat sink and the function of the Steam Generators and the major procedural transition associated with Heat Sink.

Question Source - INPO # 28135 Robinson 2 - 9/27/2004
Original Quest. KA - SE05 G2.1.27 1

5. 005 004/RO/OK/DIRECT - REPEAT/NRC EXAM 2007-4/000009 EK2.03/3.0/3.3/F/3

Given the following plant conditions on Unit 1:

- A small break LOCA is in progress.
- Only one train of SI has actuated.
- RCS Pressure is 1290 psig.
- RCS Temperature is 703°F.

In order to prevent fuel damage from inadequate core cooling, what is the reason for maintaining a secondary heat sink?

A. To provide an alternate means of RCS pressure control.

B. Reflux boiling provides the primary means of heat removal prior to voiding in the hot legs.

C. To ensure removal of RCS heat since the RCPs are expected to be running.

D✓ RCS pressure may remain so high that cooling from injection flow alone is inadequate.

ANSWER: D

A. INCORRECT. RCS Pressure is being maintained my the mass/energy balance of break flow and injection flow.
B. INCORRECT. The primary means of heat removal is the break/SI flow. The SGs are just providing a secondary heat removal function.
C. INCORRECT. SBLOCA analysis assumes that the RCPs are tripped.
D. CORRECT. Mass loss out the break is not sufficient to lower RCS pressure to a point where energy loss through the break along with injection flow is sufficient to address all decay heat removal requirements. The SG will aid in removing some of the excess decay heat.

LESSON PLAN/OBJ: RO-C-EOP02/#8, RO-C-EOP09/#9
REFERENCE: RO-C-EOP02, RO-C-EOP09

KA - 000009 EK2.03
Small Break LOCA
Knowledge of the interrelations between the small break LOCA and the following:
S/Gs
RO - 3.0 SRO - 3.3
CFR - 41.7 / 45.7
SCLR - 1B

KA Justification - Requires the knowledge for maintaining a secondary heat sink during a SB LOCA.

Question Source - Cook NRC Exam 2007-4

6. 006 005/RO/OK/MODIFIED/NRC EXAM 2004-005-1/000015 AK2.10/2.8/2.8/H/4

Unit 2 Reactor Startup is in progress with Reactor Power at 2E-8 amps and rising.

Given the following conditions on RCP 23:

- Calculated #2 Seal Leak Rate is 1.6 gpm.
- Lower Bearing water temperature is 200°F and rising.
- Motor Bearing temperature is 174°F and stable.
- Seal Leakoff temperature is 179°F and rising.
- Seal Injection Flow is 10 gpm.
- Vibrations are at 16 mils and stable.

Which ONE of the following operator actions MUST be taken based upon these conditions?

A. Manually trip the reactor, Enter 2-OHP-4023-E-0, Reactor Trip or Safety Injection, perform immediate actions, then trip the No. 3 RCP.

B.✓ Initiate reactor shutdown per 2-OHP-4021-001-003, Power Reduction and trip the No. 3 RCP after the reactor is shutdown.

C. Do NOT trip the reactor. Trip the No. 3 RCP and be in Hot Shutdown in 1 hour.

D. Do NOT trip the reactor. Trip the No. 3 RCP and close the No. 1 seal leakoff valve.

ANSWER: B

A. INCORRECT. The #2 seal is failing and will require a controlled shutdown to remove the pump from service. Reactor trip is not required.

B. CORRECT. The #2 seal is failing and will require a controlled shutdown to remove the pump from service. Reactor trip is not required, since all other parameters are within limits of the procedure and fold-out page.

C. INCORRECT. The #2 seal is failing and will require a controlled shutdown to remove the pump from service.

D. INCORRECT. The #2 seal is failing and will require a controlled shutdown to remove the pump from service.

LESSON PLAN/OBJ: RO-C-AOP-D14/#RO-C-AOP0140412-E3
REFERENCE: 02-OHP-4022-002-001, Malfunction of a Reactor Coolant Pump

KA - 000015 AK2.10
017 Reactor Coolant Pump (RCP) Malfunctions
Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions and the following:
RCP indicators and controls
RO - 2.8 SRO - 2.8
CFR - 41.7 / 45.7
SCLR - 2DR

KA Justification - Requires the knowledge of the indications and their reading that require a trip of the RCP and the associated RCS actions required prior to tripping the RCP based on these indications.

Added vibration and #2 Seal leakoff information to stem. Reduced Lower Bearing water temperature to below the manual trip requirement. Makes Answer "B" correct rather than "A".

Question Source - RO25 Audit - 5, Cook NRC Exam 2004-005-1 (#5), Audit RO24-77-7, INPO-DIRECT 20242-COOK RETAKE01

7. 007 004/RO/OK/DIRECT/MASTER 01EOPC0920-4/00WE11 EK3.1/3.3/3.9/F/3

Why does 2-OHP-4023-ECA-1.1, Loss of Emergency Coolant Recirculation, direct depressurizing the Steam Generators to 670 psig?

A. To accomplish a quick complete injection of accumulator water

B.✓ To prepare to add accumulator water sufficient to keep the core covered

C. To prevent inadvertent nitrogen addition to the RCS

D. To prevent a challenge to Reactor Coolant system integrity

ANSWER: B

A. INCORRECT. Accumulators don't inject until RCS Pressure is <650 psig. Since the RCS is assumed to be saturated RCS pressure will NOT lower to less than 670 psig.
B. CORRECT. The SGs are depressurized to lower RCS pressure and prepare for a controlled accumulator injection.
C. INCORRECT. The depressurization to 670 psig is to allow controlled Accumulator injection. Later depressurization stopped at 90 psig to prevent nitrogen injection.
D. INCORRECT. Tsat for 670 psig is ~500°F which is well above the temperatures for RCS integrity concerns. Operator may think that since the pressure is so low a challenge to inegrity may occur.

LESSON PLAN/OBJ: RO-C-EOP9/#34, 36
REFERENCE: 2-OHP-4023-ECA-1.1, 12-OHP-4023-ECA-1.1, Steps 28 &29 Background.

KA - 00WE11 EK3.1
Loss of Emergency Coolant Recirculation
Knowledge of the reasons for the following responses as they apply to the Loss of Emergency Coolant Recirculation:
Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics
RO - 3.3 SRO - 3.9
CFR - 41.5 / 41.10 / 45.6 / 45.13
SCLR - 1B/P

KA Justification - Requires the knowledge that the reason for SG Depressurization of the SGs in ECA-1.1, Loss of Emergency Coolant Recirculation is to prepare for cooling the core with accumulator injection.

Question Source - Master Bank 01EOPC0920-4
Original KA E11 EK3.2

8. 008 002/RO/OK/MODIFIED/EXAM 2004-006-2/000022 AK3.05/3.2/3.4/H/3

Unit 2 Reactor power is at 100%.

Given the following plant conditions:

- QRV-251 Charging Flow Controller is in MANUAL since Automatic control has failed.
- PRZ level is stable at program level.
- Charging and letdown are balanced.

Which ONE of the following describes the impact on the plant if a transient causes power to be lowered to 30% while 2-QRV-251 remains in MANUAL?

A. Charging flow will RISE.

B. Charging flow will LOWER.

C.✓ PRZ level will LOWER.

D. PRZ level will RISE.

ANSWER: C

A. INCORRECT. With QRV-251 Charging Flow Controller in Manual Charging flow will remain constant.
B. INCORRECT. With QRV-251 Charging Flow Controller in Manual Charging flow will remain constant.
C. CORRECT. With QRV-251 Charging Flow Controller in Manual Charging flow will remain constant. As the RCS cools down, the Pressurizer Level will lower as the water contracts.
D. INCORRECT. With QRV-251 Charging Flow Controller in Manual Charging flow will remain constant since RCS pressure is constant. As the RCS cools down, the Pressurizer Level will lower as the water contracts.

LESSON PLAN/OBJ.: RO-C-00202 / #8
REFERENCE: SOD-0202-003, Pressurizer Level Control

KA - 000022 AK3.05
Loss of Reactor Coolant Makeup
Knowledge of the reasons for the following responses as they apply to the Loss of Reactor Coolant Makeup:
Need to avoid plant transients
RO - 3.2 SRO - 3.4
CFR - 41.5 / 41.10 / 45.6 / 45.13
SCLR - 2RI

KA Justification - Requires the knowledge of the reason to avoid transients by asking the impact of a transient (power reduction) will have on Pressurizer level in the event of a malfunction of the makeup sytem to the pressuizer (charging flow control valve operating in manual)

Modified by changing power in the stem to Lower to 30% vs. Raise to 100%. Changed intial power to 100%. Changes correct answer to C vs. D.

Question Source - NRC Exam 2004-006-2, COOK02-054-1

9. 009 002/RO/OK/DIRECT/NRC EXAM 2002-023-2/000056 AK3.02/4.4/4.7/H/3

Unit 2 operators are performing 2-OHP-4023-ES-0-2, Natural Circulation Cooldown.

- 2-OHP-4023-ES-0-2, Natural Circulation Cooldown, has been entered because offsite power had been lost.
- The EDGs started and energized the AC emergency buses.
- The CRDM cooling fans cannot be manually loaded onto the AC emergency buses.
- Condensate Storage Tank water inventory is adequate for the cooldown.

Which ONE of the following describes HOW the inoperability of the CRDM fans affects the cooldown and depressurization?

A. It has no effect because the amount of RCS heat removed by running the CRDM fans is insignificant compared to the heat removed by steaming the secondary plant.

B.✓ The total upper head area cooldown rate will be less, so greater subcooling must be maintained.

C. Transition to 2-OHP-4023-ES-0-3, Natural Circulation Cooldown with Steam Void in Vessel, will be required because cooldown and depressurization will cause formation of a void in the upper head area.

D. Less subcooling should be maintained to enhance the cooldown of the upper head area.

ANSWER: B

A. INCORRECT. Does NOT address the issue of a reduced upper head cooldown rate.

B. CORRECT. 02-OHP-4023-ES-0-2, Natural Circulation Cooldown requires an RCS subcooling of 220°F in the event CRDM fans are NOT running to preclude void formation in the upper head. Normal natural circulation RCS subcooling is 86°F.

C. INCORRECT. Transition to 2-OHP-4023-ES-0-3, Natural Circulation Cooldown with Steam Void in Vessel, is NOT required given conditions which do NOT warrant an increased cooldown rate on natural circulation (i.e., CST inventory adequate for cooldown).

D. INCORRECT. The absence of the CRDM fans requires a greater RCS subcooling.

LESSON PLAN/OBJ: RO-C-EOP03 / #12 & 19
REFERENCE: 02-OHP-4023-ES-0-2, Natural Circulation Cooldown, Steps 5 & 14 and Background.

KA - 000056 AK3.02
Loss of Offsite Power
Knowledge of the reasons for the following responses as they apply to the Loss of Offsite Power:
Actions contained in EOP for loss of offsite power
RO - 4.4 SRO - 4.7
CFR - 41.5 / 41.10 / 45.6 / 45.13
SCLR - 2DR

KA Justification - Question asks for the reason that CRDM fans are desired following a Loss of Offsite Power.

Question Source: Cook NRC Exam 2002-023-2 (#17/#21), Master Bank 01EOPC0315~5
Original KA - 00WE09 - EK3.2

10. 010 038/RO/OK/NEW//000058 AA1.02/3.1/3.1/F/3

Given the following plant conditions on Unit 2:

- The unit was operating at 100%.
- A loss of 250 VDC Train A occurs.

Initially, CRID 1 and 2 power supply will be ____(1)______ aligned to _______(2)_______.

A.✓ 1) automatically

2) a 600 VAC vital bus.

B. 1) manually

2) a 600 VAC vital bus.

C. 1) automatically

2) CRP-3.

D. 1) manually

2) CRP-3.

ANSWER: A

A. CORRECT. Auto transfer will occur to the vital bus on a loss of the normal 250 VDC feed to the inverter.
B. INCORRECT. The transfer is in Automatic.
C. INCORRECT. The transfer is to 600 VAC vital bus.
D. INCORRECT. The transfer is in Automatic. The transfer is to 600 VAC vital bus.

LESSON PLAN/OBJ: RO-C-AOP-D13/RO-C-0820360101-E1
REFERENCE: 2-OHP-4024-220, Drop 29, RO-C-AOP-D13

KA - 000058 AA1.02
Loss of DC Power
Ability to operate and/or monitor the following as they apply to the Loss of DC Power: Static inverter dc input breaker, frequency meter, ac output breaker, and ground fault detector
RO - 3.1 SRO - 3.1
CFR - 41.7 / 45.5 / 45.6
SCLR - 1I

KA Justification - Requires the knowledge of the CRID Inverter (120 VAC Vital Power) response to a loss of 250 VDC and to monitor the status of the inverter following transfer.

11. 011 004/RO/OK/DIRECT/INDIANPT2003-26674/000011 EA1.07/4.4/4.4/H/3

Unit 2 was operating at 100% when a Large Break Loss of Coolant Accident occurred thirty minutes ago.

Given the following current plant conditions:

- All Control Rods are inserted.
- RCS Pressure is 40 psig.
- Pressurizer Level is 0%.
- All equipment has functioned as designed.
- RCS Subcooling is 0°F.
- Containment Radiation levels are in alarm and slowly rising.
- Containment Pressure is 4 psig and slowly lowering.

Which ONE of the following describes the status of automatic Containment Isolation?

A. Phase A was DIRECTLY actuated by a High Containment Pressure Signal, and Phase B was DIRECTLY actuated by a High-High Containment Pressure Signal.

B.✓ Phase A was DIRECTLY actuated by the Safety Injection Signal and Phase B was DIRECTLY actuated by a High-High Containment Pressure Signal.

C. Phase A was DIRECTLY actuated by the Safety Injection Signal and Phase B has NOT actuated.

D. Phase A was DIRECTLY actuated by a High Containment Pressure Signal, and Phase B has NOT actuated.

ANSWER: B

A. INCORRECT. Phase A does NOT actuate on a High Containment Pressure Signal
B. CORRECT. Phase A has actuated on the Safety Injection Signal and Phase B has actuated on High-High Containment Pressure Signal.
C. INCORRECT. Phase A Statement is correct, however Phase B should have actuated. Both Phase A and B have actuated.
D. INCORRECT. Phase B have actuated.

LESSON PLAN/OBJ: RO-C-03400/#8 and #9
REFERENCE: RQ-C-KNOW, RO-C-01100/TP28, 29, & 32

KA - 000011 EA1.07
Large Break LOCA
Ability to operate and/or monitor the following as they apply to a Large Break LOCA:
Containment isolation system
RO - 4.4 SRO - 4.4
CFR - 41.7 / 45.5 / 45.6
SCLR - 2RI

KA Justification - Requires the knowledge to determine based on monitoring whether or not a Containment Phase A/B Isolation is required (has actuated) based on indications following a Large Break LOCA.

Question Source - INPO # 26674 Indian Point 3 (Unit) - 12/11/2003

12. 012 004/RO/OK/DIRECT/NRC EXAM 2004-003-4/000008 AA1.01/4.2/4.0/F/2

Which ONE of the following describes the procedural actions in response to addressing a leaking Pressurizer (PRZ) PORV?

A.✓ 1. All PORV block valves are initially closed to lower tailpipe temperature.
2. One PORV block valve is opened at a time.
3. Leakage is determined by a rise in tailpipe temperature after each PORV block valve is re-opened.

B. 1. PORV block valves are closed one at a time.
2. Temperature on the tailpipe is monitored by the operator.
3. Leakage is determined by a lowering of tailpipe temperature after each PORV block valve is closed.

C. 1. PORV block valves are closed one at a time.
2. Temperature on the Pressurizer Relief Tank (PRT) is monitored by the operator.
3. Leakage is determined by a lowering PRT temperature after each PORV block valve is closed.

D. 1. All PORV block valves are initially closed to stabilize Pressurizer Relief Tank (PRT) temperature.
2. One PORV block valve is opened at a time.
3. Leakage is determined by a rise in PRT temperature after each PORV block valve is re-opened.

ANSWER: A

A. CORRECT. The procedure requires that all PORV Block Valves be initially closed. Once tailpipe temperature is lowering, the block valves are opened 1 at a time to check for a rise in tailpipe temperature.
B. INCORRECT. All Block Valves are initially closed.
C. INCORRECT. All Block Valves are initially closed. PRT conditions are checked but not used to determine leaky valves.
D. INCORRECT. PRT conditions are checked but not used to determine leaky valves.

LESSON PLAN/OBJ: RO-C-AOP-D2 / #RO-C-0020920412-E3
REFERENCE: OHP-4022-002-009, Leaking Pressurizer Power Operated Relief Valve

KA - 000008 AA1.01
Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
Ability to operate and/or monitor the following as they apply to the Pressurizer Vapor Space Accident:
PZR spray block valve and PORV block valve
RO - 4.2 SRO - 4.0
CFR - 41.7 / 45.5 / 45.6
SCLR - 1P

KA Justification - Questions requires ability to determine how the Block valves are manipulated and the results are moniotred during a PRZ PORV leak (vapor space leak).

Question Source - Cook NRC Exam 2004-003-4, AUDIT02-BOTH-34

13. 013 003/RO/OK/DIRECT/NRC EXAM 2002-036-1/000054 AA2.04/4.2/4.3/H/3

Unit 1 Reactor has been holding at 30% for the last hour due to Chemistry.

Given the following plant conditions:

- West Main Feedwater pump is stopped.
- East Main Feedwater pump is operating.
- AFW pumps are aligned for automatic operation.
- AMSAC is aligned in NORMAL.

The East Main Feedwater pump trips and a manual Reactor Trip is initiated.
The current conditions exist:

- Narrow Range Steam Generator levels lowered to 33%.
- Steam Dumps indicate 10% open.

Which ONE of the following statements correctly describes the AFW pump status after the Reactor Trip with no further operator actions?

A. The Motor Driven and Turbine Driven AFW pumps have NOT started.

B.✓ The Motor Driven AFW Pumps have auto started but the Turbine Driven AFW pump has NOT started.

C. The Turbine Driven AFW Pump has auto started but the Motor Driven AFW pumps have NOT started.

D. The Motor Driven and Turbine Driven AFW pumps have all auto started.

ANSWER: B

A. INCORRECT. Motor driven pumps start on loss of Main FW.
B. CORRECT. The Motor Driven AFW pumps will auto start on the loss of both Main FW pumps but the Turbine driven AFW pump will not.
C. INCORRECT. Turbine driven AFW will only start on Lo-Lo SG level, AMSAC, and RCP Bus UV and Motor Driven will start on these plus loss of Main FW, Blackout, & SI.
D. INCORRECT. Turbine driven AFW will not auto start until the AMSAC time delay is complete. AMSAC will not ARM until power has been raised to >40%.

LESSON PLAN/OBJ: RO-C-05600 / #7
REFERENCE: 01-OHP-4024-115, Drop 52; 1-OHP-4021-001-006, Power Escalation, SOD-05600-001

KA - 000054 AA2.04
Loss of Main Feedwater (MFW)
Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW):
Proper operation of AFW pumps and regulating valves
RO - 4.2 SRO - 4.3
CFR - 41.7 / 41.10 / 43.5 / 45.13
SCLR - 3PEO

KA Justification - Requires the knowledge of the auto starts of the MD and TD AFW pumps during a loss of MFP event.

Question Source - Cook NRC EXAM 2002-036-1
Slightly changed stem conditions and changed from operator actions to conditions with no operator actions.

14. 014 006/RO/OK/NEW//000065 AA2.05/3.4/4.1/F/2

Given the following plant conditions:

- An AEO has reported a major air leak in the plant air system.
- The crew is implementing 2-OHP-4022-064-001, Control Air Malfunction.

Which ONE of the following is the highest pressure which procedurally requires a manual Reactor Trip?

A. Plant Air header pressure at 79

B. Plant Air header pressure at 84

C.✓ Control Air header pressure at 79

D. Control Air header pressure at 84

ANSWER: C

A. INCORRECT. Should be control air pressure.
B. INCORRECT. Should be control air pressure. Plant Air Headers Crossties close at 85 psig. Trip requirement is less than 80 psig.
C. CORRECT. 02-OHP-4022-064-001, Control Air Malfunction directs a reactor trip if **Control Air Pressure** is <80 psig.
D. INCORRECT. Trip requirement is less than 80 psig. Plant Air Headers Crossties close at 85 psig.

LESSON PLAN/OBJ: RO-C-06401/#8
REFERENCE: 02-OHP-4022-064-001, Control Air Malfunction Step 1

KA - 000065 AA2.05
Loss of Instrument Air
Ability to determine and interpret the following as they apply to the Loss of Instrument Air:
When to commence plant shutdown if instrument air pressure is decreasing
RO - 3.4 SRO - 4.1
CFR - 41.7 / 41.10 / 43.5 / 45.13
SCLR - 1F

KA Justification - Requires the knowledge of when a Reactor Trip is required due to lowering control air header pressure.

15. 015 068/RO/OK/DIRECT/TURKEYP-2003-1215/000055 EA2.03/3.9/4.7/F/3

Given the following plant conditions on Unit 2:

- A Loss of All AC Power has occurred.
- The crew is performing the actions of 2-OHP-4023-ECA-0.0, Loss of All AC Power.
- The operators have been unable to restore power.

Which ONE of the following describes the actions required for the safeguards equipment (other than the ESW pumps) and why?

A.✓ The control switches for the safeguards equipment are placed in Pull-to-Lock to prevent a potential bus overload when power is restored.

B. The control switches for the safeguards equipment are placed in Pull-to-Lock to prevent the possibility of an uncontrolled cooldown and depressurization of the RCS when power is restored.

C. The control switches for the safeguards equipment are verified to be in automatic alignment on the 4KV bus(es) that get power back so if SI is required, it will occur without operator action.

D. The control switches for the safeguards equipment for ONE train are placed in Pull-to-Lock and the control switches for the other train are verified to be in automatic. This ensures alignment for injection without operator action.

ANSWER: A

A. CORRECT. All major loads on the bus are stripped prior to energizing the bus to prevent excessive loading as pumps initially start.
B. INCORRECT. The action is correct but the reason is incorrect.
C. INCORRECT. This action has the potential to overload the energized 4 Kv bus when power is restored consequently complicating the mitigation efforts.
D. INCORRECT. Incorrect; both trains of ECCS are placed in Pull-to-Lock.

LESSON PLAN/OBJ: RO-C-EOP14/#9
REFERENCE: RO-C-EOP14; 12-OHP-4023-ECA-0.0, Step 7 (Including Background)

KA - 000055 EA2.03
Loss of Offsite and Onsite Power (Station Blackout)
Ability to determine and interpret the following as they apply to a Station Blackout:
Actions necessary to restore power
RO - 3.9 SRO - 4.7
CFR - 41.7 / 41.10 / 43.5 / 45.13
SCLR - 1P

KA Justification - Requires the knowledge of the correct position of control board switches prior to aligning power to a bus to prevent overloading the bus when power is restored.

Question Source - TURKEY Point-2003-1215

16. 016 002/RO/OK/NEW//000077 2.4.46/4.2/4.2/H/3

Given the following plant conditions on Unit 2:

- The unit was operating at 12% power.
- Reserve Aux Transformer 201AB was supplying Buses 2A and 2B.
- Reserve Aux Transformer 201CD was supplying Buses 2C and 2D.

An electrical transient occured on the Grid.

During the transient the following values were reached:

	Electrical Bus			
Parameter	**2A**	**2B**	**2C**	**2D**
Frequency (Hz)	57	57	59.7	59.7
Local Voltage (Volts AC)	4100	4100	4200	4200
CR Indication (Volts AC)	118.2	118.2	121.1	121.1

- Ann. 207, Drop 10, RCP BUSSES UNDER FREQ TRIP is LIT.

Which ONE of the following describes the expected status of the reactor trip breakers and the Reactor Coolant Pumps?

Note: Assume no operator action.

A. Reactor Trip Breakers remain Closed
All 4 RCPs remain running.

B.✔ Reactor Trip Breakers Trip Open.
All 4 RCPs remain running.

C. Reactor Trip Breakers Trip Open.
All 4 RCPs trip off.

D. Reactor Trip Breakers remain Closed
ONLY 2 RCPs remain running.

ANSWER:B

A. INCORRECT. A Reactor trip is generated above 10% (P-7) power when 2/4 buses<58.4 Hz. (operator may confuse P-7 with P-8 blocks, P-8 associated with 2/4 RCP Low flow).
B. CORRECT. The RCPs no Longer trip on Underfrequency (Recent Plant Modification)
C. INCORRECT. A Reactor trip is generated above 10% (P-7) power when 2/4 buses<58.4 Hz.
D. INCORRECT. A Reactor trip is generated above 10% (P-7) power when 2/4 buses<58.4 Hz. (operator may confuse P-7 with P-8 blocks, P-8 associated with 2/4 RCP Low flow).

LESSON PLAN/OBJ: RO-C-01100 / #2 & 6
REFERENCE: 02-OHP-4024-207 Drop 10 RCP Busses Under Freq Trip, RO-C-01100 RPS/ESFAS Signals, RO-C-08200 TP 92 & 100

KA - 000077 2.4.46
Generator Voltage and Electric Grid Disturbances
Emergency Procedures/Plan
Ability to verify that the alarms are consistent with the plant conditions.
RO - 4.2 SRO - 4.2
CFR - 41.10 / 43.5 / 45.3 / 45.12
SCLR - 2RI

KA Justification - Question tests ability of candidate to predict the plant response to a grid disturbance based on Alarm & Indications. This demonstrates the ability to determine if the alarms are consistent with the plant conditions.

17. 017 001/RO/OK/DIRECT/MASTER-01EOPC0801-1/000038 2.4.11/4.0/4.2/F/3

The crew was operating Unit 2 at 100% when High secondary radiation and lowering PRZ level caused the crew to suspect a SG tube leak or rupture.

Which ONE of the following describes the THRESHOLD condition for initiating a reactor trip and safety injection per 02-OHP-4022-002-021, Steam Generator Tube Leak?

The THRESHOLD is when Pressurizer level can not be maintained with __(1)__ CCP(s) in service and letdown __(2)__ .

A. 1. one
   2. in service

B.✓ 1. one
   2. isolated

C. 1. two
   2. in service

D. 1. two
   2. isolated

ANSWER: B

A. INCORRECT. The procedure directs the operator to maximize charging flow and reduce/then isolate letdown. With letdown still in service the rate of leakage would be considered a Leak.

B. CORRECT. The procedure directs the operator to maximize charging flow and reduce/then isolate letdown. If PZR level can NOT be maintained in this configuration it is a rupture.

C. INCORRECT. A second charging pump is NOT started within this procedure. Plausible since 2 CCPs are allowed within 02-OHP-4022-002-020 Excessive RCS Leakage.

D. INCORRECT. A second charging pump is NOT started within this procedure. Plausible since 2 CCPs are allowed within 02-OHP-4022-002-020 Excessive RCS Leakage.

LESSON PLAN/OBJ: RO-C-EOP08/#1
REFERENCE: OHP-4022-002-021 Steam Generator Tube Leak Step 2, RO-C-EOP08 pg. 11

KA - 000038 2.4.11
Steam Generator Tube Rupture (SGTR)
Emergency Procedures/Plan
Knowledge of abnormal condition procedures.
RO - 4.0 SRO - 4.2
CFR - 41.10 / 43.5 / 45.13
SCLR - 1P

KA Justification - Requires the knowledge of the difference of the tube leak vs. rupture as defined the SG Tube Leak abnormal procedure. This also determines the entry condition for the EOPs from the AOP.

Question Source - MASTER-01EOPC0801-1

18. 018 002/RO/OK/MODIFIED/NRC EXAM 2006-062-2/000062 2.2.40/3.4/4.7/H/3

Given the following plant conditions:

- Both Units are in Mode 1.
- Engineering has determined that a relay has failed that will prevent the SI auto-start of the U1 East Essential Service Water (ESW) pump.
- All of the other U1 East ESW Pump start signals will function as designed.
- The Unit 1 East ESW discharge crosstie valves are de-energized in the open position while maintenance replaces the motors on the valves.

Which ONE of the following describes the operability and Technical Specification (TS) applicability associated with these conditions?

A. All ESW related trains are still OPERABLE because Unit 1 East ESW pump can still be manually started, and a service water TS LCO action statement would NOT be entered.

B. Only Unit 1 East ESW Train is INOPERABLE and a service water TS LCO action statement would be entered because the auto start is required to be operable.

C. All ESW related trains are still OPERABLE because the Unit 1 East ESW pump will start automatically if the discharge pressure falls below 40 psig and a service water TS LCO action statement would NOT be entered.

D.✓ Both Unit 1 East and Unit 2 West ESW Trains are INOPERABLE and the ESW TS LCO action statements for both units must be entered because the crossties are open.

ANSWER: D

A. INCORRECT. Auto start is required for operability.

B. INCORRECT. Technical Specification 3.7.8 Essential Service Water Systems, SR 3.7.8.3 requires the auto start function of the ESW pump for operability. However, with the crossties open, this required Unit 2 LCO entry as well.

C. INCORRECT. While the pump may start at 40 psig the SI auto start would also be required for operability.

D. CORRECT. If the crossties are open then the LCO action statement also applies to Unit 2.

LESSON PLAN/OBJ: RO-C-01900/#15
REFERENCE: Technical Specification 3.7.8 Essential Service Water Systems, SR 3.7.8.3, RO-C-01900 TP 20 & 22

KA - 000062 2.2.40
Loss of Nuclear Service Water
Equipment Control
Ability to apply technical specifications for a system.
RO - 3.4 SRO - 4.7
CFR - 41.10 / 43.2 / 43.5 / 45.3
SCLR - 3SPK

KA Justification - Requires the ability to determine the TS operability of the ESW system based on plant/system status provided.

Modified stem to add crosstie valves are Open. Reworded Distractors making D the Correct answer vs. B.

Question Source - Cook NRC EXAM 2006-062-2, INPO # 2836 Point Beach 1 - 8/2/1999
Original Quest. KA - 076000.G2.2

19. 019 212/RO/OK/DIRECT/NRC EXAM 2001-19307/000001 AK1.18/3.4/3.8/H/3

Given the following plant conditions:

- The plant was at 80% power and stable at EOL.
- A fault caused a continuous rod withdrawal.
- The rod motion was stopped after 20 steps by placing the Rod Control to MANUAL.
- The reactor did NOT trip.

What were the final effects of the fuel temperature and moderator temperature coefficients when the plant had stabilized?

A. The fuel temperature and moderator temperature coefficients have added positive reactivity

B.✓ The fuel temperature and moderator temperature coefficients have added negative reactivity.

C. The fuel temperature coefficient added negative reactivity and moderator temperature coefficient added positive reactivity.

D. The fuel temperature coefficient added positive reactivity and moderator temperature coefficient added negative reactivity

ANSWER: B

A. INCORRECT. Both the MTC and FTC are negative at EOL.
B. CORRECT. Both the MTC and FTC are negative at EOL.
C. INCORRECT. MTC adds negative reactivity at EOL.
D. INCORRECT. FTC adds negative reactivity at EOL.

LESSON PLAN/OBJ: RO-C-GF04/#2 & 7
REFERENCE: RO-C-GF04

KA - 000001 AK1.18
Continuous Rod Withdrawal
Knowledge of the operational implications of the following concepts as they apply to Continuous Rod Withdrawal:
Fuel temperature coefficient
RO - 3.4 SRO - 3.8
CFR - 41.8 / 41.10 / 45.3
SCLR - 2DR

KA Justification - Requires the knowledge of the relative reactivity feedback from Fuel Temperature and Moderator temperature over core life, and the effect of a rod withdrawal event on fuel temperature and moderator temperature.

Question Source - INPO Bank #19307 - COOK NRC EXAM 2001
Original Question KA - ..000001.K1.18

20. 020 045/RO/OK/DIRECT/RO24 AUDIT-027-45/00WE16 EK2.1/3.0/3.3/F/3

Given the following plant conditions:

- A Small Break LOCA occurred 12 hours ago.
- Containment pressure is 1.2 psig.
- Containment air temperature is 215ºF.
- OHP-4023-FR-Z-3, Response to High Containment Radiation Level, is entered.

Which ONE of the following verifications is a major action category of OHP-4023-FR-Z-3?

A. Both Containment Recirculation Fans are running.

B. Upper and Lower Containment Ventilation Fans are running.

C.✓ Containment Ventilation Isolation has occurred.

D. Control Room Ventilation System is in ISOLATE.

ANSWER: C

A. INCORRECT. Plausible because CEQ fans are run during a LOCA but they are run to prevent localized hydrogen concentrations and they are NOT addressed within OHP-4023-FR-Z-3.

B. INCORRECT. Plausible since these fans do provide some air filtration however they are shutdown during accident conditions.

C. CORRECT. Step 1 of the procedure ensures that CVI has occured.

D. INCORRECT. Plausible since the Isolate position is used to maintain a habitable Control Room atmosphere during accident conditions but it is not addressed in this procedure.

LESSON PLPAN/OBJ: RO-C-EOP13/#6
REFERENCE: OHP-4023-FR-Z-3, Response to High Containment Radiation Level, RO-C-EOP13 TP-56

KA - 00WE16 EK2.1
High Containment Radiation
Knowledge of the interrelations between the High Containment Radiation and the following:
Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features
RO - 3.0 SRO - 3.3
CFR - 41.7 / 45.7
SCLR - 1F

KA Justification - Requires knowledge of the procedural interrelationships between High Containment Radiation and the components and functions of a CVI.

Question Source - RO24 Audit-027-45, RO22 Audit-Both-12
OLD KA - WE 16EK3.2

21. 021 001/RO/OK/DIRECT/NRC EXAM 2002-026-1/00WE14 EK2.2/3.4/3.8/H/3

Given the following plant conditions on Unit 2:

- A LOCA occurred thirty minutes ago.
- Containment Pressure has risen to 5 psig.
- The crew has completed steps of 2-OHP-4023-ES-1.3, Transfer to Cold Leg Recirculation, to align RHR/CTS suctions to the recirculation sump and the CCP/SI suctions to RHR Discharge.
- ONLY the Train A CCP, SI, RHR, and CTS pumps are operating.
- The next step of 2-OHP-4023-ES-1.3 directs the crew to "Check if RHR Spray is Required".

Based on the indications above, which ONE of the following would best describe the required action **AND** the reason for the decision?

A. Place RHR spray in service NOW since ALL of the requirements are met.

B.✔ Place RHR spray in service ONLY after RHR has injected for 50 minutes to ensure adequate core cooling.

C. Do NOT place RHR spray in service because the RHR pump suction is NOTaligned to the RWST.

D. Do NOT place RHR spray in service because ONLY one RHR pump is operating.

ANSWER: B

A. INCORRECT. RHR has NOT injected for 50 minutes. (A LOCA occurred on Unit 2 thirty minutes ago.)
B. CORRECT. RHR is required if only 1 CTS pump is operating. After RHR has injected for 50 minutes the core is sufficiently cooled to allow RHR to be diverted to support spray functions.
C. INCORRECT. RHR spray is required 50 minutes after the accident. It is assumed that RHR will be on Recirculation at this time.
D. INCORRECT. After RHR has injected for 50 minutes the core is sufficiently cooled to allow RHR to be diverted to support spray functions.

LESSON PLAN/OBJ: RO-C-EOP13 / #13, RO-C-EOP09 / #36
REFERENCE: OHP-4023-FR-Z-1, Response To High Containment Pressure Step 4 & Background, OHP-4023-ES-1.3, Transfer to Cold Leg Recirculation Step 17 & Background

KA - 00WE14 EK2.2
High Containment Pressure
Knowledge of the interrelations between the High Containment Pressure 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 - 3.4 SRO - 3.8
CFR - 41.7 / 45.7
SCLR - 2RI

KA Justification - Requires the knowledge of the limitation for when RHR spray may be used during High Containment Pressure conditions, and the basis for the limit. These Actions are contained within ES1.3 and Z.1.

Question Source - Cook NRC Exam 2002-026-1, 01EOPC1313-2

22. 022 144/RO/OK/NEW//000028 AK3.05/3.7/4.1/H/3

Given the following plant conditions on Unit 1:

- The unit is at 100% power.
- Pressurizer Level Control selector is in the 1-2 position.
- Pressurizer Level Channel 1 (NLP-151) fails low.

Which ONE of the following describes the reason that Charging Flow and/or PZR Level control are placed in Manual in accordance with OHP-4022-013-010, Pressurizer Level Instrument Malfunction?

If NO operator action is taken following the channel failure:

A. charging flow will lower and PRZ level will lower until heaters are de-energized.
   Rx Trip will occur on OTΔT from the lowering PRZ pressure.

B. charging flow rises and PRZ level rises.
   PRZ PORVs will open as the steam space is compressed.

C. charging flow will lower and PRZ level will lower until heaters are de-energized.
   PZR pressure will lower to Rx Trip setpoint.

D.✓ charging flow rises and PRZ level rises.
   PRZ level will rise to a Hi Lvl Rx Trip setpoint.

ANSWER: D

A. INCORRECT. Charging flow will rise.
B. INCORRECT. Charging flow rises, but pressure is maintained by sprays. No PORV actuation is expected.
C. INCORRECT. Charging flow will rise.
D. CORRECT. Charging flow rises and PRZ level rises (due to charging flow and loss of letdown). PRZ level will continue to rise to a Hi Lvl Rx Trip.

LESSON PLAN/OBJ: RO-C-AOP-D3/#RO-C-AOP0340412-E1
REFERENCE: RO-C-AOP-D3

KA - 000028 AK3.05
Pressurizer (PZR) Level Control Malfunction
Knowledge of the reasons for the following responses as they apply to the Pressurizer Level Control Malfunctions:
Actions contained in EOP (AOP) for PZR level malfunction
RO - 3.7 SRO - 4.1
CFR - 41.5 / 41.10 / 45.6 / 45.13
SCLR - 3PEO

KA Justification - Requires the knowledge of the plant response to a controlling PZR level channel failing low and understanding the reason for taking manual control is to prevent a reactor trip on high PZR level (knowledge of the consequences of "NO ACTION" on the plant).

23. 023 076/RO/OK/DIRECT/KEWAUNEE2006-30436/00WE03 EA1.1/4.0/4.0/H/3

Given the following plant conditions on Unit 1:

- A Small Break LOCA has occurred.
- Offsite Power was lost on the reactor trip.
- The actions of 1-OHP-4023-ES-1.2, Post LOCA Cooldown And Depressurization, are in progress.
- Both CCPs are running with suction aligned to the RWST.
- Both RHR Pumps are stopped in Neutral.
- Both SI Pumps are running.
- The crew is ready to depressurize the RCS to refill the Pressurizer.

Which ONE of the following is the FIRST method available to the operator to commence the RCS depressurization?

The operator will open:

A.✓ One PZR PORV to vent the PZR.

B. All Pressurizer PORVs to vent the PZR.

C. The PZR Aux Spray Valve to spray down the PZR steam space.

D. One PZR Normal Spray Control valve to spray down the PZR steam space.

ANSWER: A

A. CORRECT. Since normal spray is not available (RCP Busses De-energized from loss of offsite power), the next choice is to use one PZR PORV.

B. INCORRECT. Opening MORE THAN ONE PORV is NOTan appropriate action. Only one PORV should be used to minimize the potential for a PORV sticking open.

C. INCORRECT. This action is an alternate method in the event that a PORV is not available.

D. INCORRECT. This is the "normal" method used to depressurize the RCS. However, with 4kV Buses deenergized, the RCPs are NOT running and are therefore unable to provide the driving head for normal sprays.

LESSON PLAN/OBJ: RO-C-EOP09/#36
REFERENCE: 12-OHP-4023-ES-1.2, Step 13 Background

KA - 00WE03 EA1.1
LOCA Cooldown and Depressurization
Ability to operate and/or monitor the following as they apply to the LOCA Cooldown and Depressurization:
Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features
RO - 4.0 SRO - 4.0
CFR - 41.7 / 45.5 / 45.6
SCLR - 3PEO

KA Justification - Requires the ability to depressurize the RCS during a post LOCA cooldown and depressurization and the ability to determine the depressurization method available based on plant conditions.

Question Source - INPO Bank #30436 - KEWAUNEE-222006
Original Question KA - WE03EA1.1

24. 024 096/RO/OK/DIRECT/SALEM2002-23125/000005 AA2.03/3.5/4.4/F/3

Given the following plant conditions on Unit 2:

- The unit is at 100% power
- One Control Rod in Bank D Group 1 was found stuck at 190 steps.
- While aligning the remainder of the rods in Bank D to 190 steps an additional Control Rod in Bank D Group 2 was found stuck at 210 steps.
- It has been determined that both rods are mechanically bound.

In accordance with Technical Specifications, which ONE of the following describes the action required within one hour?

A.✓ Determine that Shutdown Margin requirements are satisfied.

B. Determine that QPTR requirements are satisfied or enter the applicable action statement.

C. Verify all peaking factors are within acceptable limits.

D. Align the remainder of rods in the affected banks within 12 steps of the stuck rods.

ANSWER: A

A. CORRECT. Technical Specifications 3.1.4, Rod Group Alignment Limits, Condition A, requires SDM verification within 1 hour.
B. INCORRECT. Plausible since rod misalignment can cause skewed QPTR in the core.
C. INCORRECT. Plausible since rod misalignment can cause power peaks within the core.
D. INCORRECT. Plausible since this is TS relates to rod alignment limits.

LESSON PLAN/OBJ: RO-C-AOP-D8/#RO-C-AOP0240412-T1
REFERENCE: Tech Spec 3.1.4

KA - 000005 AA2.03
Inoperable/Stuck Control Rod
Ability to determine and interpret the following as they apply to the Inoperable/Stuck Control Rod:
Required actions if more than one rod is stuck or inoperable
RO - 3.5 SRO - 4.4
CFR - 41.7 / 41.10 / 43.5 / 45.13
SCLR - 1F

KA Justification - Requires the ability to determine the actions that are required per Tech Specs within one hour when TWO Control Rods become inoperable.

Question Source - INPO Bank #23125 - SALEM-1142002
Original Question KA - ..005.AA2.03

25. 025 004/RO/OK/DIRECT/NRC EXAM 2002-004-2/000024 AA2.01/3.8/4.1/F/3

Following a rapid power reduction from 80% power due to a Feedwater pump trip, the following plant conditions exist on Unit 1:

- Reactor power is 59%.
- Turbine Power is 620 Mwe.
- Rod Control is in MANUAL.
- All other controls in AUTO.
- Feedwater and Steam Flows are equal.
- An Emergency Boration is being performed per 1-OHP-4021-005-007, Operation Of Emergency Boration Flow Paths, due to the ROD BANK D LOW-LOW (Rod Insertion Limit) alarm being lit.

Given these conditions, which ONE of the following would be used to verify that negative reactivity is being added?

A. Tref lowering with a negative Start-up Rate.

B. Tavg lowering with approximately -0.3 dpm Start-up Rate.

C. ROD BANK D LOW-LOW alarm clearing with Tref Lowering

D✓ A flow of 45 gpm indicated on QFI-410 with Tavg Lowering

ANSWER: D

A. INCORRECT. Tref is driven by turbine first stage pressure.
B. INCORRECT. Boration will lower Tave, but the negative start-up rate is a post trip SUR.
C. INCORRECT. ROD BANK D LOW-LOW alarm clearing is only a sign that rods have moved. Other factors than Tave can cause rod motion and Tref would not be expected to lower.
D. CORRECT. Procedually flow rate is $\geq$ 44 gpm. Tavg lowering would indicate that flow is reaching the core.

LESSON PLAN/OBJ: RO-C-AOP-D2/#RO-C-AOP0200412-E3
REFERENCE: 1-OHP-4021-005-007, 1-OHP-4024-110, Drop 39, RO-C-AOP-D2 TP-11

KA - 000024 AA2.01
Emergency Boration
Ability to determine and interpret the following as they apply to the Emergency Boration:
Whether boron flow and/or MOVs are malfunctioning, from plant conditions
RO - 3.8 SRO - 4.1
CFR - 41.7 / 41.10 / 43.5 / 45.13
SCLR - 1F

KA Justification - Requires the knowledge of how to verify the proper operation of the emergency boration and what parameter in the RCS (Tave) may be used as a verification that the system is working properly.

Question Source - Cook NRC Exam 2002-004-2
Original Question KA - ..000024.K1.01

26. 026 004/RO/OK/NEW//000074 2.1.30/4.4/4.0/H/3

During the response to a LOCA, the crew implements 2-OHP-4023-FR-C-1, Response to Inadequate Core Cooling. The crew is about to depressurize the SGs to atmospheric conditions.

Given the following plant conditions:

- Containment pressure is 3.2 psig and lowering.
- A loss of CRID 1 and 2 has resulted in no power available to the SG PORV Controllers.

Which ONE of the following procedural actions will be needed to depressurize the SGs?

A. Dump steam to condensers using the steam dump system.

B.✓ Depressurize the SGs using the local SG PORV backup control stations.

C. Dump steam using the Turbine Driven AFW Steam Supply.

D. Remove control air from the SG PORVs to fail the valves open.

ANSWER: B

A. INCORRECT. Based on containment pressure. a Main Steam Isolation has occured. The Steam Dumps are not available. Additionally, a loss of CRID 2 will result in Steam Dumps NOT being available.

B. CORRECT. 2-OHP-4023-FR-C.1 dictates to use the local PORV stations in the even that the normal steam dumps and control of SG PORVs from the control room is unavailable.

C. INCORRECT. The SG pressure is relieved using the TDAFP Steam Supply valves in 2-OHP-4023-FR-H.2 when SG Pressure is high. The SG PORVs are used to provide a quicker and more controlled cooldown

D. INCORRECT. The SG PORVs fail closed on loss of air signal.

LESSON PLAN/OBJ: RO-C-EOP10/#12
REFERENCE: 2-OHP-4023-FR-C.1

KA - 000074 2.1.30
Inadequate Core Cooling
Conduct of Operations
Ability to locate and operate components, including local controls.
RO - 4.4 SRO - 4.0
CFR - 41.7 / 45.7
SCLR - 3SPK

KA Justification - Requires the ability to locate to local controls for the SG PORVs when control room control is not available.

27. 027 026/RO/OK/MODIFIED/RO25AUD-26/000076 2.1.25/3.9/4.2/H/3

Given the following plant conditions on Unit 2:

- The unit is operating at 40% power.
- Chemistry has reported the following confirmed sample results.

Dose Equivalent I-131:	125 µCi/gram
Gross Activity:	60 µCi/gram
E-bar:	1.40 Mev

Which ONE of the following describes the required actions and/or limits?

Note: Tech Spec 3.4.16, RCS Specific Activity is attached.

A.✔ The Dose Equivalent I-131 value requires that power be maintained at slightly less than 64%.

B. The Gross Activity value requires that power be maintained at slightly less than 80%.

C. The Dose Equivalent I-131 value requires the plant to be in Mode 3 with Tavg less than 500°F within 6 hours

D. The Gross Activity value requires the plant to be in Mode 3 with Tavg less than 500°F within 6 hours

ANSWER: A

A. CORRECT. 100/E-bar is 71 so gross activity is less than required. Since Dose Equivalent I-131 is > 1 µCi/gram Action A.1 applies. The required action states that LCO 3.0.4.c is applicable which allows power to be raised to the limit per the applicable figure 3.4.16-1 (~64%)

B. INCORRECT. The plant power is limited based on Dose Equivalent I-131 and Figure 3.4.16-1 NOT Gross Acitivity

C. INCORRECT. Continued operation is acceptable since I-131 is within limits of 3.4.16-1.

D. INCORRECT. 100/E-bar is 71 so gross activity is less than required.

LESSON PLAN/OBJ: RO-C-0200\#10
REFERENCE: TS 3.4.16 RCS Specific Activity.

Attachment Provided: Tech Spec 3.4.16, RCS Specific Activity

KA - 000076 2.1.25
High Reactor Coolant Activity
Conduct of Operations
Ability to interpret reference materials, such as graphs, curves, tables, etc.
RO - 3.9 SRO - 4.2
CFR - 41.10 / 43.5 / 45.12
SCLR - 3SPR

KA Justification - Requires the ability to use TS 3.4.16 (RCS Activity) graph for I-131 to determine TS actions.

Modified stem and distractors to ask basic actions based on graph and values.

Question Source - RO25 Audit-26, Audit RO22-BOTH-70 (#62)

28. 028 003/RO/OK/MODIFIED/NRC EXAM 2004-017-4/064000 K1.02/3.1/3.6/H/3

Given the following sequence of events:

- Unit 1 and Unit 2 were operating at 100% power.
- ALL of the Unit 1 and Unit 2 Essential Service Water (ESW) pumps were operating with the Unit Crossties CLOSED.
- Unit 2 tripped due to a turbine Electro-Hydraulic Control fluid leak.
- Unit 1 remained on line.
- The Unit 2 Reserve Transformers are unavailable.
- Both Unit 2 Emergency Diesel Generators (EDGs) started and loaded; however, Bus T21D failed to energize.

Assuming NO operator actions have been taken yet, which ONE of the following describes the ESW cooling water status for the Unit 2 EDGs and the required actions, if any?

A.✓ 2CD EDG must be tripped as ESW cooling has been lost since the Unit Crossties are closed.

B. 2CD EDG has ESW cooling supplied by the Unit 2 West ESW Pump since the Alternate ESW supply automatically aligned.

C. 2AB EDG must be tripped as ESW cooling has been lost since the Unit Crossties are closed.

D. 2CD EDG has ESW cooling supplied by the Unit 1 West ESW Pump since the Unit Crossties automatically opened.

ANSWER: A

A. CORRECT. When bus T21D is lost the Unit 2 East ESW Pump Trips, this will cause a low header pressure condition. The Unit 1 West ESW pump would supply ESW Cooling water if the crossties were open but they do NOT automatically open.

B. INCORRECT. The 2CD Diesel Generator could be supplied if the alternate ESW supply was manually opened (recent change).

C. INCORRECT. Diesel Generator 2AB has cooling from the auto start of the Unit 2 West ESW Pump (Would also have cooling from the Unit 1 East ESW).

D. INCORRECT. When bus T21D is lost the Unit 2 East ESW Pump Trips, this will cause a low header pressure condition. The Unit 1 West ESW pump will NOT supply cooling since the crossties are closed. Candidate may assume crossties open since the cross unit pumps start on an SI.

LESSON PLAN/OBJ: RO-C-03201/#4, RO-C-01900/3.f
REFERENCE: SOD-01900-001, Essential Service Water

KA - 064000 K1.02
Emergency Diesel Generator (ED/G) System
Knowledge of the physical connections and/or cause-effect relationships between the ED/G System and the following systems:
ED/G cooling water system
RO - 3.1 SRO - 3.6
CFR - 41.2 to 41.9 / 45.7 to 45.8
SCLR - 3SPK

KA Justification - Requires the knowledge of the interrelationships between the ESW cooling flowpath and the EDGs and the knowledge of the crosstie capability from the opposite unit.

Question Source - COOK NRC Exam 2004-017-4, COOK NRC Exam 2002-095-1

MODIFIED - Changed to Unit crossties Closed with ALL 4 pumps running in stem which changed distractor A to the correct answer (vs. D). Added reasons to the distractors.

29. 029 003/RO/OK/DIRECT/RO23 AUDIT 074-3/008000 K1.03/2.8/3.0/F/3

Given the following plant conditions on Unit 2:

- The East CCW HX is in service with the West CCW Pump running.
- CCW Surge Tank level is stable.
- CRS-4301, East CCW HX Radiation Monitor, generates an External Failure Alarm due to a faulty flow switch.

Which ONE of the following describes the response of the CCW system for the given conditions?

A. No automatic actions will occur since the West CCW pump is running

B. No automatic actions will occur since the CRS-4401, West CCW HX Radiation Monitor is still functioning.

C. 2-CMO-420, West CCW HX Outlet, opens and 2-CMO-410, East CCW HX Outlet, closes

D✓ 2-CRV-412, CCW Surge Tank Vent Valve, will automatically close.

ANSWER: D

A. INCORRECT. Plausible since the East CCW HX monitor may be associated with the East pump but either radiation monitor will cause the surge tank vent to isolate.

B. INCORRECT. Plausible since the east monitor has failed and the west is still operational but an alarm or failure of either will cause the Surge tank vent to isolate.

C. INCORRECT. Plausible if assumption is made that CCW will automatically align to the opposite train from the rad monitor.

D. CORRECT. 2-CRV-412, CCW Surge Tank Vent Shutoff Valve closes on High Rad Level Alarm, Low Sample Flow, External Failure on CRS-4300/4400, Channel 4301 - East and/or Channel 4401-West.

LESSON PLAN/OBJ: RO-C-01350/#4
REFERENCE: 12-OHP-4024-139, Drop #29

KA - 008000 K1.03
Component Cooling Water System (CCWS)
Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems:
PRMS
RO - 2.8 SRO - 3.0
CFR - 41.2 to 41.9 / 45.7 to 45.8
SCLR - 1I

KA Justification - Requires the knowledge of the cause and affect relationships between the CCW system and the Process Rad Monitoring system following an external failure of the rad monitor.

Question Source - RO23 AUDIT 074-3, Similar to Cook NRC Exam 2006-084-11 (SRO question that also requires alignment & Operability call)

30. 030 330/RO/OK/NEW//078000 K1.05/3.4/3.5/H/3

Given the following plant conditions on Unit 1:

- The unit was operating at 100% power.
- A Loss of Control Air has occurred.
- Control Air Header is reading 0 psig.

Which ONE of the following describes the effect of the loss of control air on the Main Steam Isolation Valves (MSIVs).

The MSIVs:

A. will remain open and can be closed using the hydraulic unit.

B. will remain open and can be closed by locally aligning nitrogen to one MSIV dump valve.

C.✓ close due to MSIV dump valves failing open on loss of air.

D. close due to loss of control air to the hydraulic unit.

ANSWER: C

A. INCORRECT. MSIVs close due to loss of air to the dump valves.
B. INCORRECT. MSIVs close due to loss of air to the dump valves. There is no backup nitrogen supply to the dump valves. Plausible since PORVs have nitrogen backup.
C. CORRECT. MSIV dump dumps fail open on loss of air causing the MSIV to close on steam delta-p on the steam actuating piston.
D. INCORRECT. The hydraulic unit is comprised of spring and motor operated components. Loss of air has no effect on the hydraulic unit.

LESSON PLAN/OBJ: RO-C-05103/#8
REFERENCE: SOD-05103-003

KA - 078000 K1.05
Instrument Air System (IAS)
Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems:
MSIV air
RO - 3.4 SRO - 3.5
CFR - 41.2 to 41.9 / 45.7 to 45.8
SCLR - 2RI

KA Justification - Requires the knowledge of the effect that a loss of control air will have on the Main Steam Islolation Valves.

31. 031 024/RO/OK/DIRECT/RO24 AUDIT 014-24/063000 K2.01/2.9/3.1/F/3

Unit 2 was operating at about 100% power when a Complete Loss of Onsite and Offsite AC power occurred 3 hours ago. Unit 2 dispatched operators after 30 minutes to shed the large Non-Essential DC loads.

Given the following plant conditions:

- The crew transitioned to 2-OHP-4023-ECA-0-0, Loss Of All AC Power, and stabilized SG pressures.
- Power has just been restored from Emergency Power.
- While performing Step 30, power could NOT be restored to the battery chargers for the N train, 2AB, and 2CD 250VDC buses. (The actions of step 30 to restore 600V AC Busses and Control Room Cooling were successfully performed.)

Which ONE of the following describes the impact of failing to restore the battery chargers?

When the Batteries completely discharge:

A. all AFW flow will be lost when the AFW pump discharge valves fail closed

B.✓ the ability to start and stop ECCS pumps from the control room will be lost.

C. the Emergency Power feed breaker will trip open resulting in another Loss of AC.

D. all vital instrumentation will be lost.

ANSWER: B

A. INCORRECT. Plausible since many AOVs will fail closed on loss of DC and the TDAFW Pump Valves use DC but will fail as is while the MDAFW valves may be operated.

B. CORRECT. A loss of DC control power will prevent breaker operations with the control switch (and trip functions).

C. INCORRECT. Plausible since the breaker is affected by loss of DC but this breaker will not open from a loss of DC - it must receive a trip signal (overload).

D. INCORRECT. Plausible since the (CRIDs) vital instruments are normally supplied from inverters that have DC backup but will remain powered from AC backup sources.

LESSON PLAN/OBJ: RO-C-08201/#4
REFERENCE: RO-C-08201, Engineered Safety System Electrical Distribution System

KA - 063000 K2.01
D.C. Electrical Distribution System
Knowledge of bus power supplies to the following:
Major DC loads
RO - 2.9 SRO - 3.1
CFR - 41.7
SCLR - 1I

KA Justification - Requires the knowledge of the effect of a loss of DC will have on major loads (breaker control power) and ability of those breakers to reposition.

Question Source - RO24 AUDIT 014-24, COOK NRC Exam 2002-074, RO23 Audit-27

32. 032 002/RO/OK/DIRECT/NRC EXAM 2004-022-2/005000 K2.01/3.0/3.2/H/3

Unit 1 was operating at 100% power when a Large Break LOCA occurred. The West RHR pump tripped on overload and can NOT be restarted.

SI has been reset and the crew has just stopped both Emergency Diesel Generators and returned to standby in accordance with 1-OHP-4023-E-1, Loss of Reactor or Secondary Coolant.

The following sequence of events occurs:

- The T11D, 4kV AC ESF Bus subsequently loses normal power but is re-energized by the 1CD Emergency Diesel Generator.
- T11D Automatic load sequencing is complete.

Which ONE of the following statements correctly describes the status the East RHR Pump?

The East RHR Pump:

A. has tripped and automatically restarted.

B. was NOT affected by the loss of Bus T11D.

C.✓ has tripped and may be manually started immediately.

D. has tripped and may NOT be manually started until the load conservation signal has been reset.

ANSWER: C

A. INCORRECT. The pumps will NOT automatically start following a load shed.
B. INCORRECT. The East RHR is powered from T11D.
C. CORRECT. The ECCS timer and SI signals are the only auto starts for the RHR Pumps. The pumps will NOT automatically start following a load shed. The East RHR is powered from T11D while the West RHR is powered from T11A.
D. INCORRECT. The RHR Pump is not prevented from starting due to a load shed.

LESSON PLAN/OBJ: RO-C-01700/#8
REFERENCE: RO-C-01700, 1-OHP-4023-E-1, Step 8

KA - 005000 K2.01
Residual Heat Removal System (RHRS)
Knowledge of bus power supplies to the following:
RHR pumps
RO - 3.0 SRO - 3.2
CFR - 41.7
SCLR - 3PEO

KA Justification - Requires the knowledge of the bus power supply for the East RHR pump on loss of power with EDG start and load of bus.

Question Source - Cook NRC EXAM 2004-022-2

33. 033 004/RO/OK/MODIFIED/NRC EXAM 2004-051-3/062000 K3.02/4.1/4.4/H/3

Unit 1 is in Mode 3. The 4160 VAC distribution system is being supplied by the Reserve Auxiliary Transformers (RATs). Due to a system disturbance, indicated voltage on the safeguards busses drops.

Given the following plant conditions:

- T11A Voltage Indication is 112 Volts.
- T11B Voltage Indication is 114 Volts.
- T11C Voltage Indication is 113 Volts.
- T11D Voltage Indication is 112 Volts.

Which ONE of the following describes the FINAL plant response if voltage remains at these values for an extended period?

A.✓ ALL safeguards busses will be energized by their respective EDG.

B. ONLY T11A and T11D busses will be energized by their respective EDG.

C. ONLY T11C and T11D bus will be energized by its respective EDG.

D. ONLY T11A, T11C, and T11D busses will be energized by its respective EDG.

ANSWER: A

A. CORRECT. An Undervoltage condition of 113 V will energize 62-1 T11A & T11D After a 111 Second delay it will open ALL the T -bus feeders breakers will open causing them to lose power. This will cause the EDGs to start and energize ALL of the busses from the EDGs.

B. INCORRECT. T11B & T11C will deenergize since T11A & T11D are < 113V

C. INCORRECT. An Undervoltage condition of 113 V will energize 62-1 T11A. After a 111 Second delay it will open T11A9 and T11B1 causing T11A and T11B to lose power. This will cause the EDG to start and energize T11A and T11B. Plausible if candidate assumes that both T11A and T11 B must be < 114V

D. INCORRECT. T11B will also recieve a trip signal and be energized by the EDG.

LESSON PLAN/OBJ: RO-C-08201/#6
REFERENCE: RO-C-08201, Engineered Safety Systems Electrical pg. 30-31, SD-08201 pg. 25-26 and Figure 1; Annunciator #121 Response, Drop 78; RC-C-KNOW

KA - 062000 K3.02
A.C. Electrical Distribution System
Knowledge of the effect that a loss or malfunction of the A.C. Distribution System will have on the following:
ED/G
RO - 4.1 SRO - 4.4
CFR - 41.7 / 45.6
SCLR - 3PEO

KA Justification - Requires the knowledge of the effect of a loss of AC (lowering RATS voltage) will have on the EDG supply to the Emergency Bus.

Question Source - Cook NRC EXAM 2004-051-3

Modified to change stem with T11D voltage at 112 vs. 114. This makes A the correct answer. Answers B,C,&D changed to plausible distractors.

34. 034 005/RO/OK/DIRECT/RO24 AUDIT 032-5/007000 K3.01/3.3/3.6/H/3

Which ONE of the following describes the effects of continued power operation with a leaking pressurizer PORV? (Assuming NO operator actions.)

A. There are NO adverse effects. The PRT is designed to handle continuous in-leakage.

B. The cyclic temperature stresses in combination with inner wall erosion on the PORV tailpipe may lead to premature piping failure.

C.✓ The PRT rupture disc may break with subsequent elevated radiation, temperature and pressure indications in containment.

D. Mechanical breakdown of the PORV seating surface may cause the PORV to fail when needed for overpressure protection.

ANSWER: C

A. INCORRECT. Plausible since the PRT can handle in-leakage but the PRT is not designed for continuous input without any actions to cool and drain.

B. INCORRECT. Plausible since cyclic temperatures could lead to excessive stresses but, with a constant leak the temperatures will not be cycling, PORV seat cutting/erosion may be a concern but not inner wall erosion.

C. CORRECT. The tank design is based on the requirement to condense and cool a discharge of pressurizer steam equal to 110 percent of the volume above the 100%-power pressurizer water level set-point. If the temperature in the tank rises above 126°F during plant operation, the tank is cooled by spraying in cool water and draining out the warm mixture to the Waste Disposal System. The tank is not designed to accept a continuous discharge from the pressurizer.

D. INCORRECT. Plausible since damage may occur (PORV seating may erode) but it would be available for overpressure protection.

LESSON PLAN/OBJ: RO-C-00202/#7
REFERENCE: UFSAR Chapter: 4 Page 18, RO-C-00202, SOD-00202-001

KA - 007000 K3.01
Pressurizer Relief Tank/Quench Tank System (PRTS)
Knowledge of the effect that a loss or malfunction of the PRTS will have on the following:
Containment
RO - 3.3 SRO - 3.6
CFR - 41.7 / 45.6
SCLR - 2RI

KA Justification - Requires the knowledge of a adverse effects of the PRT rupture disc failure and the effect this will have on the containment atmosphere.

Question Source - RO24 AUDIT 032-5, Cook NRC Exam 2002-99-1 (83/80)

35. 035 005/RO/OK/MODIFIED/HARRIS2004-27486/022000 K3.02/3.0/3.3/H/3

Given the following plant conditions:

- Unit 2 is at 100% power.
- Lower Containment Cooling NESW supply is throttled to all ventilation units.
- A failure of a Lower Containment Cooling Supply regulator results in full flow through the cooling coils.
- Average containment temperature lowers from 119°F to 109°F.
- Charging Flow Control is in MANUAL
- Assume RCS Pressure and Temperature remain Constant.

Which ONE of the following describes the change in indicated Pressurizer level due to this lowering in Containment temperature?

Density rising in the _______ leg causes indicated pressurizer level to read ________ than actual level.

A. reference; higher

B.✓ reference; lower

C. variable; higher

D. variable; lower

ANSWER: B

A. INCORRECT. Indicated level will be less than actual level.

B. CORRECT. Pressurizer Level uses a wet reference leg DP level indicator. This compares the pressure of the full reference leg with the pressure of the actual water in the pressurizer. When these are equal the level indicates 100%. As the temperature in Containment and therefore the reference leg lowers the density & weight of the reference leg rises. This means that the level in the pressurizer will indicate lower for the same initial actual level.

C. INCORRECT. Indicated level will be less than actual level. Reference leg density rises.

D. INCORRECT. Reference Leg density rises.

LESSON PLAN/OBJ: RO-C-GF27/9d
REFERENCE: RO-C-GF27, Sensors and Detectors pg. 51 & 52

KA - 022000 K3.02
Containment Cooling System (CCS)
Knowledge of the effect that a loss or malfunction of the CCS will have on the following:
Containment instrumentation readings
RO - 3.0 SRO - 3.3
CFR - 41.7 / 45.6
SCLR - 3SPK

KA Justification - Requires the knowledge of the effect a malfunction of the containment cooling system will have on the the pressurizer level instruments located in containment.

Modified Stem to make Containment Cooling malfunction that Lowered temperature vs. Raise. Changed Correct Answer to Distractor B vs. A.

Question Source - INPO # 27486 Harris 1 - 3/24/2004, Similar to Cook NRC Exam -2006-045-5 : INPO # 26772 Kewaunee, Unit 1 - 2/2/2004

36. 036 283/RO/OK/DIRECT/MASTER 01002C0102-1/003000 K4.07/3.2/3.4/F/2

Which ONE of the following describes the leakoff flow of the RCP #2 seal during normal operations?

A.✓ 3 gph to the Reactor Coolant Drain Tank via RCP standpipe.

B. 2 gpm to the Volume Control Tank.

C. 100 cc/hr to the Reactor Coolant Pump standpipe.

D. 100 cc/hr directly to the Reactor Coolant Drain Tank.

ANSWER: A

A. CORRECT. RCP #2 Seal Leakoff flowpath is directed to the #2 Seal Standpipe, which flows through and orifice to the RCDT.
B. INCORRECT. RCP #1 seal leakoff is directed to the VCT at a design of 3 gpm.
C. INCORRECT. Flowrate is for the #3 seal leakoff which vents to atmosphere and RCDT.
D. INCORRECT. Flowrate is for the #3 seal leakoff which vents to atmosphere and RCDT.

LESSON PLAN/OBJ: RO-C-00201/#3
REFERENCE: SD-00201; SOD-00201-001

KA - 003000 K4.07
Reactor Coolant Pump System (RCPS)
Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following:
Minimizing RCS leakage (mechanical seals)
RO - 3.2 SRO - 3.4
CFR - 41.7
SCLR - 1I

KA Justification - Requires the knowledge of the RCP Seal Leakoff flowpath for the No. 2 seal used to minimize RCS leakage.

Question Source - MASTER Bank 01002C0102-1

37. 037 006/RO/OK/NEW//026000 K4.07/3.8/4.1/F/3

Which ONE of the following sets of parameters lists the indications/design features provided for CTS pump protection when pump suction is aligned to the containment recirc sump?

1) Minimum Recirculation Level lights
2) Recirc Sump Level Low Alarm
3) Low Suction Pressure CTS Pump Trip
4) Low Sump Level CTS Pump Trip
5) Low CTS Pump Suction Pressure Alarm
6) Low Sump Level Status Light

A.✓ 1, 2, 6

B. 1, 3, 5

C. 2, 3, 4

D. 4, 5, 6

ANSWER: A

A. CORRECT. Indications of adequate level in the containment recirc sump include minimum level status lights, low recirc sump level alarm, and associated status lights.

B. INCORRECT. There are no low suction pressure alarms or trips for the CTS pumps. Plausible since the RHR pumps have a low pressure alarm and a low RWST level trip.

C. INCORRECT. There are no low suction pressure or low level CTS pump trips.

D. INCORRECT. There is no low level CTS pump trip. There is no low CTS pump suction pressure alarm.

LESSON PLAN/OBJ: RO-C-03400/#9
REFERENCE: RO-C-03400

KA - 026000 K4.07
Containment Spray System (CSS)
Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following:
Adequate level in containment sump for suction (interlock)
RO - 3.8 SRO - 4.1
CFR - 41.7
SCLR - 1F

KA Justification - Requires the knowledge of the design features that provide CTS pump protection when suction is aligned to the containment recirc sump.

38. 038 001/RO/OK/NEW//010000 K4.02/3.0/3.4/F/2

Which ONE of the following sets contains ONLY conditions that will cause the UNIT 1 pressurizer heaters to automatically deenergize?

1. Pressurizer Level Control Channel - 5% below program level.
2. Pressurizer Level Control Channel - less than 17% level.
3. Pressurizer Level Bistable Channel - less than 17% level.
4. Pressurizer Level Cold Calibration Channel - less than 17% level.

A. 1 and 3

B.✓ 2 and 3

C. 2 and 4

D. 1 and 4

ANSWER: B

A. INCORRECT. Protection is provided by the control channel but at 17%. Plausible since at 5% above program the heaters energize.

B. CORRECT. Protection is provided when either the CONTROL or BISTABLE channel is <17%

C. INCORRECT. The Interlock does NOT exist on the cold cal channel - plausible since this instrument is used for main control as the RCS is cooled down and it reads lower than hot cals (hot cal channels don't provide true protection at lower temperatures.

D. INCORRECT. Protection is provided by the control channel but at 17%. Plausible since at 5% above program the heaters energize. The Interlock does NOT exist on the cold cal channel - plausible since this instrument is used for main control as the RCS is cooled down and it reads lower than hot cals (hot cal channels don't provide true protection at lower temperatures.

LESSON PLAN/OBJ: RO-C-00202/#5, #12
REFERENCE: SOD-00202-003, RO-C-00202 TP-16

KA - 010000 K4.02
Pressurizer Pressure Control System (PZR PCS)
Knowledge of PZR PCS design feature(s) and/or interlock(s) which provide for the following:
Prevention of uncovering PZR heaters
RO - 3.0 SRO - 3.4
CFR - 41.7
SCLR - 1P

KA Justification - Requires the knowledge of the heater protection interlocks provided by the PZR level control system.

39. 039 008/RO/OK/DIRECT/RO24 AUDIT 052-8/073000 A1.01/3.2/3.5/F/3

With an event in progress, noble gas readings are rising rapidly on the Unit Vent Radiation Monitor VRS-1500.

Which ONE of the following directly shifts the sample air flow through VRS-1500 to the emergency lineup (opens 1-VRV-317, Bypass Valve and closes 1-VRV-318, Monitor Valve)?

High alarm on:

A. VRS-1503, Unit Vent Effluent Iodine Gas Radiation Monitor.

B. VRS-1505, Unit Vent Effluent Low Range Noble Gas Radiation Monitor.

C. VRS-1506, Unit Vent Effluent Area Radiation Monitor.

D✓ VRS-1509, Unit Vent Effluent High Range Noble Gas Radiation Monitor.

ANSWER: D

A. INCORRECT. Plausible because VRS-1505 - Low range noble gas does have an automatic action but it Closes12-RRV-306.
B. INCORRECT. Plausible because VRS-1505 - Low range noble gas does have an automatic action but it Closes12-RRV-306 on HIGH alarm.
C. INCORRECT. Plausible since this monitor does monitor the flowpath but has alarm actions only.
D. CORRECT. High alarm on Unit vent effluent high range noble gas radiation monitor VRS-1509 will Close1-VRV-318 and Open 1-VRV-317.

LESSON PLAN/OBJ: RO-C-01350/#4d
REFERENCE: OHP 4024.139 DROP 5

KA - 073000 A1.01
Process Radiation Monitoring (PRM) System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM System controls including:
Radiation levels
RO - 3.2 SRO - 3.5
CFR - 41.5 / 45.5
SCLR - 1F

KA Justification - Requires the knowledge of the operation of Rad Monitor VRS-1500 that ensure monitoring is maintained to prevent exceeding design limits at elevated rad levels.

Question Source - RO24 AUDIT 052-8, Master Bank 01013C50XX-1

40. 040 036/RO/OK/DIRECT/RO25 AUDIT- 36/013000 K5.02/2.9/3.3/H/3

Given the following plant conditions:

- Unit 2 is at 100% power.
- Containment pressure instrument Channel #3, 2-PPP-301, declared inoperable.
- Required actions per 2-OHP-4022-013-011 Containment Instrumentation Malfunction have been completed.

CRID 2 has just lost power.

Which ONE of the following statements explains the impact on the Safety Injection system and expected operator actions?

A. No Safety Injection logic is satisfied. Implement 2-OHP-4021-082-008, Operation of CRID Power Supplies, to address the CRID Failure.

B. Only Train "A" safety injection actuation logic was satisfied, but equipment will NOT automatically actuate. Implement 2-OHP-4023-E-0, Reactor Trip or Safety Injection, and inform the operators that they must individually reposition/start the equipment.

C.✓ Train "A" and "B" safety injection actuation logic were satisfied. Implement 2-OHP-4023-E-0, Reactor Trip or Safety Injection. Both Trains equipment will automatically actuate.

D. Train "A" and "B" safety injection actuation logic were satisfied, but Train "A" equipment will NOT automatically actuate. Implement 2-OHP-4023-E-0, Reactor Trip or Safety Injection, and inform the operators that they must individually reposition/start each Train "A" component.

ANSWER: C

A. INCORRECT. SI will Actuate.
B. INCORRECT. Both Trains of SI will Actuate (All channels input into both trains)
C. CORRECT. Channels 2, 3, & 4 of Containment Pressure cause an SI. This signal will result from the bistables being tripped on Channel 3 & the Loss of Power to Channel 2 bistables. The SI signal is actuated on Both SI trains. Train A SSPS output is powered from CRID 1 & Train B is powered from CRID 4 so all actuations will occur automatically.
D. INCORRECT. Both Trains output relays still have power.

LESSON PLAN/OBJ: RO-C-AOP-D1/#RO-C-AOP0350412-E1; RO-C-01100/#6, ;
RO-C-AOP-D13/RO-C-0820360101-E1
REFERENCE: RO-C-01100, TP25; 02-OHP-4022-013-011; RO-C-AOP-D13, TP17

KA - 013000 K5.02
Engineered Safety Features Actuation System (ESFAS)
Knowledge of the operational implications of the following concepts as they apply to the ESFAS:
Safety system logic and reliability
RO - 2.9 SRO - 3.3
CFR - 41.5 / 45.7
SCLR - 3PEO, 2RI

KA Justification - Requires the knowledge of the redundancy of inputs into the ESFAS system and the ability to determine if coincidence is made up for actuation of the system.

Question Source - RO25 AUDIT- 36

41. 041 293/RO/OK/NEW//061000 K6.02/2.6/2.7/F/2

Given the following plant conditions on Unit 1:

- The unit is at 100% power when a Reactor Trip and SI occurs.
- While verifying AFW flow the operator notes the following:
  - The West MDAFP has tripped on overload.
  - The TDAFP has failed to start.
  - The East MDAFP total flow is 250 x 10^3 pph.

Which ONE of the following describes the capabilities of the East AFW Pump with regards to the minimum AFW flow for RCS Decay Heat removal and the SGs being supplied from this pump?

The East MDAFP will:

A. supply flow to the 1 and 4 SGs. This flow meets the minimum required for RCS Decay Heat Removal.

B.✓ supply flow to the 2 and 3 SGs. This flow meets the minimum required for RCS Decay Heat Removal.

C. supply flow to the 1 and 4 SGs. This flow alone does NOT meet the minimum required for RCS Decay Heat Removal.

D. supply flow to the 2 and 3 SGs. This flow alone does NOT meet the minimum required for RCS Decay Heat Removal.

ANSWER: B

A. INCORRECT. Capacity is ~240 x 10^3 pph, but it is aligned to automatically feed 2 and 3 SGs.
B. CORRECT. Minimum flow required for initial decay heat removal is 240K pph and the East MDAFP is aligned to automatically feed 2 and 3 SGs.
C. INCORRECT. Flow rate is adequate for decay heat removal, and the East feeds 2 and 3 SGs.
D. INCORRECT. Flow rate is adequate for decay heat removal.

LESSON PLAN/OBJ: RO-C-05600/#2 & 3
REFERENCE: RO-C-05600 Auxiliary Feedwater System pg. 10-11, TP 37

KA - 061000 K6.02
Auxiliary / Emergency Feedwater (AFW) System
Knowledge of the effect of a loss or malfunction of the following will have on the AFW System components:
Pumps
RO - 2.6 SRO - 2.7
CFR - 41.7 / 45.7
SCLR - 1B

KA Justification - Question asks candidate to identify the AFW flow delivered to the generators from a single AFW pump following the loss of the other 2 AFW Pumps and the ability of the single pump to satisfy design requirements.

42. 042 006/RO/OK/MODIFIED/NRC EXAM 2004-069-5/006000 K6.05/3.0/3.5/F/3

Unit 2 has experienced a loss of both CCW pumps in Mode 3.

Given the following plant conditions:

- NEITHER Unit 2 CCW pump can be restarted.
- Unit 2 East CCPs is running
- Unit 2 West CCP is in standby.
- 2-OHP-4022-016-004, Loss of Component Cooling Water, is in progress.

Which ONE of the following describes the procedural requirements for CCP operation based on these conditions?

A. Immediately stop the East CCP until the Unit 1 CCW system has been crosstied.

B. Run the East CCP for up to 5 minutes and then stop the East CCP and initiate a CVCS crosstie to Unit 1.

C✓ Run the East CCP as long as it continues to operate or until a CVCS crosstie to Unit 1 can be performed.

D. Run the East CCP as long as it continues to operate and then start the West CCP at minimum flow from the RWST until the Unit 1 CCW system has been crosstied.

ANSWER: C

A. INCORRECT. One pump should be run as long as possible to allow time to align CVCS crosstie.

B. INCORRECT. One pump should be run as long as possible to allow time to align CVCS crosstie. (The pump may trip after 1.5 minutes)

C. CORRECT. 02-OHP-4022-016-004 has a note prior to step 4 that describes the possible damage that may occur to a CCP on the loss of CCW. The note and procedure directs that one CCP be saved until CCW is restored. The other pump should be run as long as possible to allow time to align CVCS crosstie

D. INCORRECT. One pump should be run as long as possible to allow time to align CVCS crosstie. (The pump may trip after 1.5 minutes) The running pump is aligned to the RWST with minimum flow and then Unit 1 crosstie for CCW is established.

LESSON PLAN/OBJ: RO-C-AOP-D14/RO-C-AOP0560412-E2
REFERENCE: 02-OHP-4022-016-004, Loss of Component Cooling Water

KA - 006000 K6.05
Emergency Core Cooling System (ECCS)
Knowledge of the effect of a loss or malfunction of the following will have on the ECCS:
HPI/LPI cooling water
RO - 3.0 SRO - 3.5
CFR - 41.7 / 45.7
SCLR - 1P

KA Justification - Question Tests knowledge of action required (effect) for a loss of CCW cooling to the CCP (HPSI)

Question Source - Cook NRC Exam 2004-069-5, AOP1CAOP5.13

Modified Stem to change from Both pumps running to the West in Standby. Changed all distractors to add CVCS or CCW crosstie. Changed actions in All distractors based on 1 pump in standby.

43. 043 007/RO/OK/DIRECT/NRC EXAM 2004-047-1/039000 K5.05/2.7/3/1/F/3

Given the following plant conditions:

- The plant is in Mode 3 performing a cooldown in preparation for a refueling outage.
- A malfunction of the Steam Generator Power Operated Relief Valves causes the cooldown rate to exceed Technical Specification limits.

Which ONE of the following actions is required per TECHNICAL SPECIFICATIONS and why?

A. Restore cooldown rate limits within 1 hour to provide adequate margin from ductile failure of the reactor vessel.

B. Immediately stop any further cooldown. Maintain temperature for 6 hours to allow temperature stabilization throughout the reactor vessel wall.

C. Stop cooldown within 15 minutes. Maintain temperature for 12 hours to allow temperature stabilization throughout the reactor vessel wall.

D.✓ Restore cooldown rate limits within 30 minutes to provide adequate margin from brittle failure of the reactor vessel.

ANSWER: D

A. INCORRECT. Time to restore is 30 minutes. Concern is brittle failure.
B. INCORRECT. Time to restore is 30 minutes. Plausible since soak time would aid the situation, this is NOT a required action and the time is excessive.
C. INCORRECT. Time to restore is 30 minutes. Plausible since soak time would aid the situation, this is NOT a required action and the time is excessive.
D. CORRECT. Technical Specification 3.4.3 requires the RCS temperature to be restored to within Limits in 30 minutes. The concern of excessive cooldown rates to brittle failure caused by the tensile stresses on the inner wall.

Note: All times in distractors are plausible times based on other TS actions.

LESSON PLAN/OBJ: RO-C-GF23/#14
REFERENCE: Technical Specification 3.4.3 Pressure/Temperature Limits & Basis B3.4.3; RO-C-GF23 Brittle Fracture and Vessel Thermal Stress

KA - 039000 K5.05
Main and Reheat Steam System (MRSS)
Knowledge of the operational implications of the following concepts as they apply to the MRSS:
Bases for RCS cooldown limits
RO - 2.7 SRO - 3.1
CFR - 41.5 / 45.7
SCLR - 1B, 1F

KA Justification - Question asks for reasons (fundamental knowledge) of RCS cooldown limits and actions required if they are exceeded (operational impact).

Modified stem to add TS and Distractors B & C.

Question Source - RO24 Audit-043-7 Update ITS (SWP 7-28-05), NRC Exam 2004-047-1, INPO-DIRECT 20886-PALISADES01

44. 044 058/RO/OK/DIRECT/NRC EXAM 2004-062-3/103000 A1.01/3.7/4.1/F/2

Given the following plant conditions on Unit 2:

- The unit is operating at 100% power.
- A small instrument air leak inside Containment causes a slow rise in Containment pressure.
- Containment pressure is currently 0.19 psig.

In order to ensure that adequate margin to Containment Technical Specification pressure limits is maintained, which ONE of the following indicates the appropriate action to reduce Containment pressure ?

A. Maximize NESW cooling to the Containment Ventilation Units

B. Lower pressure in containment using the Containment Purge System.

C.✓ Vent containment using the Containment Pressure Relief system.

D. Verify all Upper/Lower Containment Ventilation Fans (CUV/CLV) are running.

ANSWER: C

A. INCORRECT. Increasing cooling (lowering temperature) may cause a slight pressure reduction but with continued in-leakage, a pressure release will have to be performed.
B. INCORRECT. The Containment Purge system is used only for shutdown conditions.
C. CORRECT. With the Containment Pressure rising due to air line leakage, the only way to reduce pressure is to purge air from Containment. This is accomplished with the Containment Pressure Relief System.
D. INCORRECT. Increasing cooling (lowering temperature) may cause a slight pressure reduction but with continued in-leakage, a pressure release (vent) will have to be performed.

LESSON PLAN/OBJ:RO-C-02800/#2
REFERENCE: RO-C-02800, Containment Ventilation System pg. 8,
12-OHP-4023-ES-1.1 Step 4 background.

KA - 103000 A1.01
Containment System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Containment System controls including:
Containment pressure, temperature, and humidity
RO - 3.7 SRO - 4.1
CFR - 41.5 / 45.5
SCLR - 1F

KA Justification - Question asks for operator to predict which system of control will reduce Containment Pressure to help prevent exceeding the design pressure.

Question Source: NRC Exam 2004-062-3, 21601-KEWAUNNE02

45. 045 001/RO/OK/MODIFIED/NRC EXAM 2004-049-1/059000 A1.07/2.5/2.6/H/3

Unit 2 is at 76% power with all control systems in AUTOMATIC.

Which ONE of the following describes the plant response to a trip of the East Main Feed Pump?

Note: Assume no operator action.

A. 1. The Turbine will NOT automatically runback causing SG water levels to start lowering resulting in the Feedwater Regulating Valves opening further.
2. A lower feedwater header pressure causes the West MFP speed to rise until it trips on overspeed.
3. Both the Steam-driven and Motor-driven Auxiliary Feedwater Pumps start when SG levels reach the Low-Low level setpoint.

B. 1. The Turbine will automatically runback.
2. The SG pressures will rise due to the runback requiring the West MFP speed to rise.
3. As the MFP speed rises, the MFP Suction Pressure will lower causing the Low Pressure Heater Bypass CRV-224 to open and the Middle Heater Drain Pump to start.

C. 1. The Turbine will NOT automatically runback.
2. The West MFP transfers to Speed Control with the Speed set to Maximum but will not maintain SG levels.
3. Both the Steam-driven and Motor-driven Auxiliary Feedwater Pumps start when SG levels reach the Low-Low level setpoint.

D.✓ 1. The Turbine will automatically runback.
2. The West MFP transfers to Speed Control with the Speed set to Maximum.
3. NO automatic pump starts occur.

ANSWER: D

A. INCORRECT. A turbine runback is automatically initiated. The Main FW pumps can supply 60% flow and so the West FW pump would not trip on overspeed.
B. INCORRECT. The Heater Drain pump will not auto start. The LP Heater Bypass may Open on lowering suction pressure.
C. INCORRECT. A turbine runback is automatically initiated. The Main FW pumps can supply 60% flow and so a Low-Low level would not be reached and AFW will not start.
D. CORRECT. On the loss of the East Main FW Pump a turbine runback is automatically initiated, reduced flow will cause the FW regulating valves will open further as the SGs try to maintain normal level & FW flow matched to steam flow. FW Pump transfers to Speed Control with the Speed set to Maximum

LESSON PLAN/OBJ: RO-C-05501/#RO-C-05501-E8, RO-C-AOP-D14 / #RO-C-AOP0450412-E1
REFERENCE: RO-C-05501, Main Feedwater Turbine Controls,
02-OHP-4022-055-001, Loss Of One Main Feed Pump

KA - 059000 A1.07
Main Feedwater (MFW) System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW System controls including:
Feed Pump speed, including normal control speed for ICS
RO - 2.5 SRO - 2.6
CFR - 41.5 / 45.5
SCLR - 2RI

KA Justification - Question requires candidate to predict/monitor the Automatic FW Pump Speed response to a FW pump trip.

Modified question to a higher power level (60-76%) and added turbine runback to distractors and tranfer to speed control (Recent Modification to DCS controls)

Question Source - Cook NRC Exam 2004-049-1, COOK02-069-1

46. 046 021/RO/OK/NEW//012000 A2.07/3.2/3.7/H/3

Unit 2 is operating at 100% Power when the following alarm actuates:

- Ann. 210, Drop 50, REACTOR PROT TRAIN B TROUBLE.

Investigation reveals the Train B General Warning Lamp Lit and one of the 48 VDC power supplies on Train B tripped.

Which ONE of the following describes the impact, if any, of this failure?

A. The Train B Reactor Trip breaker Shunt trip is unavailable.

B. NONE of Train B SSPS equipment will automatically actuate when required.

C. A portion of the Train B SSPS equipment will NOT automatically actuate when required.

D✓ ALL of Train B SSPS equipment will automatically actuate when required.

ANSWER: D

A. INCORRECT. The Shunt trips are powered from 250 VDC.
B. INCORRECT. The 48 VDC power supplies are 100% redundant within the SSPS train. This would be true for a loss of CRID IV and ALL 48VDC (Master relays would not acuate).
C. INCORRECT. The 48 VDC power supplies are 100% redundant within the SSPS train. This would be true if they shared the loads (Master relays would not acuate).
D. CORRECT. The 48 VDC power supplies are 100% redundant within the SSPS train. The RPS/SSPS will still function as designed but the redundancy is compromised.

LESSON PLAN/OBJ: RO-C-01101/#5 & 9
REFERENCE: RO-C-01101 Solid State Protection System, 02-OHP-4024-210 Drop 50

KA - 012000 A2.07
Reactor Protection System
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 dc control power
RO - 3.2 SRO - 3.7
CFR - 41.5 / 43.5 / 45.3 / 45.13
SCLR - 2RI

KA Justification - The questions tests the ability to determine the impact of the failure of a loss of DC to the RPS/SSPS and the required actions.

47. 047 086/RO/OK/NEW//076000 A2.01/3.5/3.7/H/3

Given the following plant conditions:

- **Unit 2** was operating at 100% power.
- The Unit 2 North NESW Pump was running.
- The Unit 2 South NESW Pump was stopped and in AUTO.

A large steamline break inside containment occurred.
- **Unit 2** offsite power is lost on the trip.
- Both **Unit 2** Diesel Generators started and are supplying their respective busses.
- **Unit 2** Containment Pressure is 3.2 psig.
- **UNIT 1** has remained operating at 100% power throughout the event.

Which ONE of the following describes the response of the NESW pumps and the required actions; if any, to restore Unit 2 NESW?

A. The Safety Injection Sequencer will start both Unit 2 NESW pumps after the appropriate time delay.

B. The Safety Injection Sequencer will start the North NESW Pump after the appropriate time delay. The South NESW Pump will start on Low System Pressure if the North NESW Pump fails to restart.

C. NESW Load Conservation will prevent the start of both Unit 2 NESW pumps. Either pump may be started by placing the control switch directly to Run after the Containment Spray Pumps have been stopped.

D✓ NESW Load Conservation will prevent the start of both Unit 2 NESW pumps. Either pump may be started by placing the control switch to Trip/Lockout and then to Run after at least 75 seconds have elapsed since the time of the Containment Spray Signal.

ANSWER: D

A. INCORRECT. The NESW pumps normally auto start (37 seconds after Load Shed) on the SI Sequencer but are locked out when a CTS signal also exists.

B. INCORRECT. The North NESW Pump would have tripped on a Load Shed and the South NESW will normally start on a Low Pressure signal, but both are locked out since a CTS signal also exists. The Control Switches must be placed in Trip/Lockout prior to run to clear the signal.

C. INCORRECT. NESW Load Conservation has prevented the pumps from starting (Tripped the pumps), but the Control Switches must be placed in Trip/Lockout prior to run to clear the signal.

D. CORRECT. NESW Load Conservation actuates on a CTS with Load Shed (or EDG on Bus). This prevents the NESW pumps from starting for 75 seconds or until the CTS signal is cleared. The Control Switch will need to be placed in Trip/Lockout to clear the signal.

LESSON PLAN/OBJ: RO-C-02000/#12
REFERENCE: RO-C-02000 Non Essential Service Water System

KA - 076000 A2.01
Service Water System (SWS)
Ability to (a) predict the impacts of the following malfunctions or operations on the SWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Loss of SWS
RO - 3.5 SRO - 3.7
CFR - 41.5 / 43.5 / 45.3 / 45.13
SCLR - 2RI

KA Justification - The Question tests knowledge of the reason for the loss of the NESW (SWS) system and the required actions (procedures) to restore the pumps.

48. 048 002/RO/OK/DIRECT/NRC EXAM 2004-060-2/025000 A2.04/3.0/3.2/H/3

Given the following plant conditions on Unit 1:

- The unit was operating at 100% power when an INADVERTENT Phase A Containment Isolation occurred on Train A.
- The Crew has reset Phase A Containment Isolation and attempted to restore Control Air to Containment.
- The Control Air Containment Isolation Valves could not be opened.

Which ONE of the following describes short-term impact of the loss of air on the restoration efforts of the crew and the required compensatory actions?

A. RCP NESW Motor Air cooling water can NOT be restored. Trip the reactor and stop 3 RCPs. Perform a containment pressure relief.

B.✓ Glycol Cooling to the ice condenser can NOT be restored. Stop all Unit 1 Air Handling Units (AHUs). Monitor ice bed temperatures to ensure they remain at an acceptable level.

C. RCS overpressure protection has been lost (PORVs will NOT open). Begin a reactor shutdown and be in Mode 3 within 6 hours.

D. RCP Seal Injection is available but Seal Return can NOT be restored. Drain the PRT as required to maintain an acceptable level.

ANSWER: B

A. INCORRECT. NESW to RCP Motor Cooling valves are located outside containment and close on a Phase B Isolation. Actions are correct for Loss of NESW.

B. CORRECT. Glycol Cooling inside Containment Isolation valves VCR-11 and VCR-21 will NOT open. The AHU's are stopped if the glycol system is shutdown for more than 30 minutes. Technical Specifications requires that temperatures are maintained < 27°F.

C. INCORRECT. PORVs NRV-152 and NRV-153 have local reservoirs. Technical Specifications require a shutdown if all PORVS are lost.

D. INCORRECT. RCP Seal Injection is not isolated and Seal Return QCM-250 and QCM-350 are motor operated valves. RCP Seal Leakoff valves QRV-10, 20, 30, and 40 are fail open. Seal Return would go to the PRT if the Containment Isolation valves were closed.

LESSON PLAN/OBJ: RO-C-01000/#7
REFERENCE: RO-C-01000 Ice Condenser, TS 3.6.11, 12-OHP-4024-135 Drop 51

KA - 025000 A2.04
Ice Condenser System
Ability to (a) predict the impacts of the following malfunctions or operations on the Ice Condenser System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Containment isolation
RO - 3.0 SRO - 3.2
CFR - 41.5 / 43.5 / 45.3 / 45.13
SCLR - 2RI

KA Justification - Question tests knowledge of the impact an inadvertent Phase A Containment Isolation has on the Ice Condenser cooling system & subsequent actions.

Modified by asking for actions in stem and in all Distractors.

Question Source - Cook NRC Exam 2004-060-2

49. 049 173/RO/OK/NEW//004000 A3.02/3.6/3.6/F/3

Given the following plant conditions:

- Reactor Power is 95%.
- The East CCP pump trips.
- The West CCP is in Standby.

Which ONE of the following describes the effect of these conditions on the CVCS System operation assuming NO operation action?

A. QRV-251 (CCP Discharge Flow Control valve) throttles to minimum flow.

B.✓ QRV-160, 161, and 162 (Letdown Orifice Isolation valves) fully close.

C. CRV-470 (Letdown Temperature Control valve) throttles in the open direction.

D. QRV-111 and 112 (RC Letdown to Regen Hx valves) fully close.

ANSWER: B

A. INCORRECT. When the Charging Pumps trips Letdown will isolate but since the Pump is also lost, seal leakoff causes PRZ level to lower causing QRV-251 to open.

B. CORRECT. The Letdown Orifice Isolation valves are directly interlocked to close (isolates letdown) on a loss of the CCP.

C. INCORRECT. The Letdown temperature control valve will close since letdown flow is lost. This would be true for a loss of Regen HX CVCS flow (QRV-251 Closed) or if letdown was NOT isolated.

D. INCORRECT. The Letdown **ORIFICE** Isolation valves close, NOT the **NORMAL** Letdown Isolation Valves.

LESSON PLAN/OBJ: RO-C-00300/#14
REFERENCE: RO-C-00300, SOD-00300-001

KA - 004000 A3.02
Chemical and Volume Control System (CVCS)
Ability to monitor automatic operation of the CVCS, including:
Letdown isolation
RO - 3.6 SRO - 3.6
CFR - 41.7 / 45.5
SCLR - 1I

KA Justification - Question tests knowledge of how a CCP pump trip causes an automatic letdown isolation (closure of the orifice isolation valves).

50. 050 002/RO/OK/MODIFIED/MASTER 01055C0013-6/059000 A3.04/2.5/2.6/F/3

Given the following plant conditions:

- Unit 2 is at 55%
- Both Main Feedwater Pumps are in operation.

Which ONE of the following conditions will DIRECTLY cause a trip of the East Main Feedwater Pump (MFP)?

A. Main Turbine Trip

B.✓ Loss of MFP Turbine Condenser Vacuum

C. East MFP Turbine Lube Oil Temperature High

D. East MFP Turbine Thrust Bearing Temperature High

ANSWER: B

A. INCORRECT. A turbine trip will cause a reactor trip, which will then trip the operating MFPs. There is no DIRECT MFP trip from the turbine trip.
B. CORRECT. Condenser vacuum of $\leq$15.0 in Hg will cause a Triconix Trip of the operating MFP.
C. INCORRECT. MFP Turbine Lube Oil Temperature is an alarm function only. Plausible since there is a trip from lube Oil Pressure.
D. INCORRECT. MFP Turbine thrust Bearing Temperature High is an alarm function only. There is no DIRECT trip from this condition. Plausible since a Thrust Bearing Wear of $\pm$32 mils will cause a MFP turbine trip.

LESSON PLAN/OBJ: RO-C-05501/#RO-C-05501-E9, RO-C-05500 / #6
REFERENCE: RO-C-05500 Main FW System, RO-C-05501 Main Feed Pump Turbine Controls

KA - 059000 A3.04
Main Feedwater (MFW) System
Ability to monitor automatic operation of the MFW System, including:
Turbine driven feed pump
RO - 2.5 SRO - 2.6
CFR - 41.7 / 45.5
SCLR - 2RI

KA Justification - Question tests ability of candidate to monitor for conditions that would cause an automatic trip of the turbine driven Main Feedwater Pumps.

Question Source - Cook MASTER Bank 01055C0013-6

51. 051 001/RO/OK/DIRECT/NRC EXAM 2002-19/013000 A4.02/4.3/4.4/H/3

A Safety Injection (SI) occurred 20 minutes ago due to a large break LOCA, with a failure of reactor trip breaker "B" to open.

Which ONE of the following describes the SI signal response after pushing (then releasing) both SI reset pushbuttons?

	SI STATUS	
	Train A	**Train B**
A.✔	Reset	NOT Reset
B.	Reset	Reset
C.	NOT Reset	NOT Reset
D.	NOT Reset	Reset

ANSWER: A

A. CORRECT. With a failure of Train B reactor Trip Breaker to open, a P-4 signal is not generated on Train B. With a Large Break LOCA the SI signal will still be present preventing Train B from being reset. The SI reset and P-4 block features are train specific.
B. INCORRECT. Train B will not reset.
C. INCORRECT. Train A will reset.
D. INCORRECT. Train A will reset and Train B will not Reset.

LESSON PLAN/OBJ: RO-C-01100/#4 & #10
REFERENCE:RO-C-01100, SSPS Hardware, SOD-1100-001, SOD-1100-002

KA - 013000 A4.02
Engineered Safety Features Actuation System (ESFAS)
Ability to manually operate and/or monitor in the control room:
Reset of ESFAS channels
RO - 4.3 SRO - 4.4
CFR - 41.7 / 45.5 to 45.8
SCLR - 2RI

KA Justification - Question tests ability of the candidate to determine which SI channels will reset (monitor reset).

Original Question - COOK NRC EXAM 2002-19, RO25 Audit-01, Master bank 01011C0018,

52. 052 070/RO/OK/DIRECT/RO22 AUDIT-BOTH-69/012000 A4.06/4.3/4.3/F/3

Which ONE of the following describes the effect on the reactor trip breakers of actuating the manual REACTOR TRIP switches?

Shunt trip relays de-energize, causing the reactor trip breaker trip coils to __(1)__ ,
AND
Reactor trip breaker undervoltage coils to __(2)__ .

	(1)	(2)
A.	energize	energize
B.	de-energize	de-energize
C.✓	energize	de-energize
D.	de-energize	energize

ANSWER: C

A. INCORRECT. Manual trip de-energizes the undervoltage trip coil.
B. INCORRECT. Manual trip energizes the reactor trip breaker trip coil.
C. CORRECT. Manual trip de-energizes the undervoltage trip coil and energizes the reactor trip breakertrip coil.
D. INCORRECT. Manual trip de-energizes the undervoltage trip coil and energizes the reactor trip breaker trip coil.

LESSON PLAN/OBJ: RO-C-01101 / #2
REFERENCE: RO-C-01101

KA - 012000 A4.06
Reactor Protection System
Ability to manually operate and/or monitor in the control room:
Reactor trip breakers
RO - 4.3 SRO - 4.3
CFR - 41.7 / 45.5 to 45.8
SCLR - 1F, 1I

KA Justification - Question tests ability of operator to determine the effects on the trip breakers (monitor) from a manual actuation (operate).

Question Source - RO22 AUDIT-BOTH-69

53. 053 001/RO/OK/NEW//022000 2.1.27/3.9/4.0/F/3

Which ONE of the following describes the Technical Specification Limits and the ESFAS response associated with the **UPPER** Containment Ventilation fans?

	Upper Containment temperature limits	Fans Trip from
A.	≥60°F and ≤120°F	Phase A actuation
B.	≥60°F and ≤120°F	Phase B actuation
C.✓	≥60°F and ≤100°F	Phase A actuation
D.	≥60°F and ≤100°F	Phase B actuation

ANSWER: C

A. INCORRECT. These are the temperature limits for lower Containment.
B. INCORRECT. These are the temperature limits for lower Containment and the response of the Lower Containment Fans.
C. CORRECT. Technical Specifications Limits upper Containment to ≥60°F and ≤ 100°F. The fans trip on a Phase A Actuation.
D. INCORRECT. This is the correct limits but the response of the Lower Containment Fans.

LESSON PLAN/OBJ: RO-C-02800/#2, 9, & 11
REFERENCE: RO-C-02800, Containment Ventilation System, Technical Specification 3.6.5

KA - 022000 2.1.27
Containment Cooling System (CCS)
Conduct of Operations
Knowledge of system purpose and/or function.
RO - 3.9 SRO - 4.0
CFR - 41.7
SCLR - 1B

KA Justification - Question tests knowledge of the system purpose (maintain temperature within limits) and the function on a ESFAS signal.

54. 054 039/RO/OK/NEW/NEW/025000 2.2.12/3.7/4.1/F/3

Given the following plant conditions on Unit 2:

- Unit 2 is Operating at 100% power.
- All Containment Ice Consenser equipment/instrumentation is OPERABLE.

Which ONE of the following methods is used to ensure compliance with Tech Spec Suveillance Requirement SR 3.6.12.1, "Verify all inlet doors are closed" every 12 hours?

The Operator is required to verify:

A.✓ ONLY that Annunciator Pannel 222 Drop 84, ICE CONDENSER INLET DR POSIT alarm is NOT Lit.

B. ONLY that the ice bed temperature is less than or equal to 27°F.

C. that visual inspection of local Ice Condenser Inlet doors has been completed AND that Annunciator Pannel 222 Drop 84, ICE CONDENSER INLET DR POSIT alarm is NOT Lit.

D. ONLY that visual inspection of local Ice Condenser Inlet doors has been completed.

ANSWER: A

A. CORRECT. 02-OHL-4030-SOM-041 U2 CR M1 & 2 Shift checks requires that the alarm be clear.
B. INCORRECT. This action is to verify Ice Bed Operability and may be used if the door is Inoperable.
C. INCORRECT. A visual inspection is required weekly for the Intermediate deck doors but NOT for the Inlet doors
D. INCORRECT. A visual inspection is required weekly for the Intermediate deck doors.

LESSON PLAN/OBJ: RO-C-01000/#12
REFERENCE: TS 3.6.12 pg. 3.6.12-1, 2,&3, 02-OHL-4030-SOM-041 U2 CR M1 & 2 Shift checks pg. 111
SR 3.6.12.1 Verify all inlet doors are closed.

KA - 025000 2.2.12
Ice Condenser System
Equipment Control
Knowledge of surveillance procedures.
RO - 3.7 SRO - 4.1
CFR - 41.10 / 45.13
SCLR - 1P

KA Justification - Question exams knowledge of the surveillance requirements (which determine operability) for the Ice Condenser Inlet Doors.

55. 055 001/RO/OK/NEW//039000 2.4.20/3.8/4.3/H/3

Given the following plant conditions on Unit 2:

- A tube rupture has occurred in the SG#23.
- A Manual Reactor Trip and Safety Injection were performed.
- RCS Pressure is currently 2035 psig.
- RCS Tave is 549°F.
- SG#23 has been isolated in accordance with procedure 2-OHP-4023-E-3, Steam Generator Tube Rupture.

The Unit Supervisor has directed you to perform the cooldown to 475°F using the Steam Dumps.

Which ONE of the following describes how the cooldown would be performed and the reason?

A. Place the Steam Dumps in Steam Pressure mode and fully open them to obtain a cooldown at the maximum rate. The Steamline Break SI, Main Steam Isolation will NOT occur since SI is already actuated

B.✓ Place the Steam Dumps in Steam Pressure mode and open them in a controlled manner to prevent causing a Steamline Isolation. Block the Steamline Break SI, Main Steam Isolation as temperature lowers.

C. Block the Steamline Break SI, Main Steam Isolation and then place the Steam Dumps in Steam Pressure mode and fully open them to obtain a cooldown at the maximum rate.

D. Block the Steamline Break SI, Main Steam Isolation and then place the Steam Dumps in Steam Pressure mode and open them in a controlled manner to prevent exceeding steam flows of 1.42 x10^6 pph and causing a Steamline Isolation.

ANSWER: B

A. INCORRECT. The note before step 7 warns that a steam line isolation may occur if the dumps are opened rapidly.

B. CORRECT. The note before step 7 warns that a steam line isolation may occur if the dumps are opened rapidly. The dumps should be opened in a controlled manner and to obtain the maximum cooldown rate and then the Steam Line SI can be blocked when P-12 (541°F) is reached.

C. INCORRECT. The Steam Line SI can NOT be blocked until P-12 (541°F) is reached. (Operator may confuse P-12 setpoint of 541°F with Low Tave of 554°F)

D. INCORRECT. The Steam Line SI can NOT be blocked until P-12 (541°F) is reached. The steam flows of 1.42 x 10^6 pph will cause an Isolation until when P-12 (541°F) is reached.

LESSON PLAN/OBJ: RO-C-EOP08/#16 & 20
REFERENCE: 02-OHP-4023-E-3 Step 7 and Background, RO-C-EOP08

KA - 039000 2.4.20
Main and Reheat Steam System (MRSS)
Emergency Procedures/Plan
Knowledge of operational implications of EOP warnings, cautions, and notes.
RO - 3.8 SRO - 4.3
CFR - 41.10 / 43.5 / 45.13
SCLR - 2RI

KA Justification - Question tests knowledge of EOP note regarding operation of the Steam Dumps and Main Steam Line Isolation.

56. 056 004/RO/OK/NEW//079000 K1.01/3.0/3.1/F/3

Given the following plant conditions:

- Unit 2 Plant Air Compressor (PAC) is running.
- Unit 1 Plant Air Compressor (PAC) is out of service for maintenace.
- Both Units' Control Air Compressors (CACs) are stopped and in AUTO.
- A large leak occurs on the piping connecting the Unit 1 Plant Air Receiver to the Unit 1 side of the plant air header
- Plant Air Header pressure is slowly lowering.

Which ONE of the following statements describes the method for isolating the leak and the Control Air Supply System status following isolation?

Note: Assume leak isolation occurs prior reaching 98 psig Plant Air Header Pressure.

A. 1) Close PRV-10 **OR** PRV-11, Plant Air Header Crosstie Valves to Unit 2
2) Control Air for both units will be supplied from the Unit 2 PAC

B✓ 1) Close PRV-10 **AND** PRV-11, Plant Air Header Crosstie Valves to Unit 2
2) Control Air for both units will be supplied from the Unit 2 PAC

C. 1) Close PRV-10 **AND** PRV-11, Plant Air Header Crosstie Valves to Unit 2
2) Unit 1 Control Air will be supplied by the Unit 1 Control Air Compressor **AND** Unit 2 Control Air remains supplied from the Unit 2 PAC

D. 1) Close PRV-10 **OR** PRV-11, Plant Air Header Crosstie Valves to Unit 2
2) Unit 1 Control Air will be supplied by the Unit 1 Control Air Compressor **AND** Unit 2 Control Air will be supplied from the Unit 2 CAC

ANSWER: B

A. INCORRECT. Both PRV-10 AND PRV-11 must be closed to isolate the Unit 1 header.
B. CORRECT. Both PRV-10 AND PRV-11 must be closed to isolate the Unit 1 header. Both control air supplies come downstream of the header crosstie valves, so U2 PAC will continue to supply both units' control air headers.
C. INCORRECT. Both control air supplies come downstream of the header crosstie valves, so U2 PAC will continue to supply both units' control air headers.
D. INCORRECT. Both PRV-10 AND PRV-11 must be closed to isolate the Unit 1 header. Both control air supplies come downstream of the header crosstie valves, so U2 PAC will continue to supply both units' control air headers.

LESSON PLAN/OBJ: RO-C-06401/#3
REFERENCE: SOD-06401-002, Plant Air System

KA - 079000 K1.01
Station Air System (SAS)
Knowledge of the physical connections and/or cause-effect relationships between the SAS and the following systems:
IAS
RO - 3.0 SRO - 3.1
CFR - 41.2 to 41.9 / 45.7 to 45.8
SCLR - 1S

KA Justification - Question tests knowledge of interrelations for the piping supplies between the plant air (SAS) and the control air systems (IAS).

57. 057 002/RO/OK/NEW//011000 K2.01/3.1/3.2/F/2

Which ONE of the following describes the power supply for the Unit 1 West CCP during normal plant operation?

A.✓ Bus T11A

B. Bus T11B

C. Bus T11C

D. Bus T11D

ANSWER: A

A. CORRECT. The West CCP is a Train B pump which is supplied by the T11A Bus.
B. INCORRECT. This is a Train B bus but it does NOT supply the pumps.
C. INCORRECT. This is a Train A bus and it does NOT supply the pumps.
D. INCORRECT. The East CCP and Train A pumps are supplied from T11D.

LESSON PLAN/OBJ: RO-C-00300/#9
REFERENCE: RO-C-00300 Chemical Volume Control System

KA - 011000 K2.01
Pressurizer Level Control System (PZR LCS)
Knowledge of bus power supplies to the following:
Charging pumps
RO - 3.1 SRO - 3.2
CFR - 41.7
SCLR - 1F

KA Justification - Question tests knowledge of which bus supplies power to the West CCP which is used as part of the Pzr Level Control System.

Question Source - New

58. 058 058/RO/OK/DIRECT/RO25 AUDIT-58/055000 K3.01/2.5/2.7/H/3

Unit 2 was operating at 100% power when main condenser vacuum dropped from 28 inches vacuum to 25 inches vacuum and stabilized.

Given the following plant conditions:

- All 4 condenser steam air ejectors (SJAE) are in service.
- The system lineup is in the normal configuration.

Which ONE of the following conditions describes the cause of this problem?

A. Loss of a running Hotwell Pump's seal water supply.

B. Loss of the Main Steam supply to the SJAEs.

C.✓ The SJAE condenser drains were CLOSED.

D. The SJAE condenser drains were OPENED.

ANSWER: C

A. INCORRECT. Loss of hotwell pump seal water will only result in minor air inleakage from the Hotwell Pumps but it is well within SJAE capacity.
B. INCORRECT. Main steam does NOT supply the SJAEs. There are physical connections for main steam to be the back up supply for aux steam - but these connections are NOT used.
C. CORRECT. Removes the ability of SJAEs to remove noncondensibles and causes vacuum to lower.
D. INCORRECT. SJAE drains are normally open to allow condensate to drain out of the air ejectors. If the drain is left closed, the condensate will build up and flood out the SJAE, which could cause a loss of vacuum. This is the reverse of the distracter.

LESSON PLAN/OBJ: RO-C-GF14/#7, RO-C-05400/#3
REFERENCE: 2-OHP-4024-DCS-MT Drops 134-136, SOD- 05400-001

KA - 055000 K3.01
Condenser Air Removal System (CARS)
Knowledge of the effect that a loss or malfunction of the CARS will have on the following:
Main condenser
RO - 2.5 SRO - 2.7
CFR - 41.7 / 45.6
SCLR - 2RI

KA Justification - Question provides the effect (low vacuum) on the main condenser and asks for the cause (malfunction of SJAE/CARS)

Question Source - RO25 Audit-58, AUDIT RO23 RO-22

59. 059 001/RO/OK/NEW//045000 K4.34/2.7/2.9/H/3

Rod Control was in AUTO with Unit 1 power at 79% when MPC-253, Turbine Impulse Pressure Channel 1, failed low.

Which ONE of the following describes the expected response of the rod control system?

A. Rods will initially move in due to power mismatch. AUTO rod control will then withdraw rods to restore temperature.

B. Rods will move in due to power mismatch and due to a temperature error until placed in MANUAL. AUTO rod control may be restored after MPC-253 has been defeated and the associated bistables have been tripped.

C. Rods will initially move in due to power mismatch but will stop as the temperature error becomes large enough. AUTO rod control is NOT available until after Conditional C-5 bistable has been tripped.

D✓ Rods will move in due to power mismatch and due to a temperature error until placed in MANUAL. AUTO rod control is NOT available even after MPC-253 bistables have been tripped.

ANSWER: D

A. INCORRECT. Rods inserted due to power mismatch and temperature. (Plausible if candidate assumes MPC-254 supplies Tref - instead of MPC-253).
B. INCORRECT. The Tref function is not selectable. (Plausible if candidate assumes Tref is selectable).
C. INCORRECT. Rods inserted due to power mismatch and temperature. (Plausible if candidate assumes MPC-254 supplies Tref - instead of MPC-253).
D. CORRECT. The failure of MPC-253 would have caused the control Rods to Insert (Tref lowered) since it feeds Tref. MPC-253 also feed the 15% C-5 interlock which prevents Auto Rod Withdrawal. Since the Tref function is not selectable - Auto Rod Control is not available.

LESSON PLAN/OBJ: RO-C-01200 / #6; RO-C-AOP-D2/RO-C-AOP0390412-E2
REFERENCE: RO-C-01200 Rod Control and Rod Position Indicating System,
02-OHP-4022-013-016, TURBINE FIRST STAGE IMPULSE
PRESSURE INSTRUMENT MALFUNCTION

KA - 045000 K4.34
Main Turbine Generator (MT/G) System
Knowledge of MT/G System design feature(s) and/or interlock(s) which provide for the following:
Operation of CRDS in manual mode at T/G power below 15%
RO - 2.7 SRO - 2.9
CFR - 41.7
SCLR - 2RI

KA Justification - Question tests knowledge of the impact of the transmitter which feeds the C-5 interlock failing.

60. 060 001/RO/OK/DIRECT/NRC EXAM 2004 091-1/071000 K5.04/2.5/3.1/F/3

The In-Service Waste Gas Decay Tank #2 has a HYDROGEN concentration of 4.8%. Per TRM 8.7.12, Explosive Gas Mixture, which ONE of the following is the **HIGHEST** OXYGEN concentration allowed in Gas Decay Tank #2?

A. 2.0 %

B.✓ 3.0 %

C. 4.0 %

D. 5.0 %

ANSWER: B

A. INCORRECT. 3.0% is the Highest allowed.
B. CORRECT. TRM 8.7.12, Explosive Gas Mixture, TRO 8.7.12, requires Oxygen Concentration of less than or equal to 3% when Hydrogen concentration is > 4%
C. INCORRECT. Only if H2 <4%
D. INCORRECT. Only if H2 <4%

LESSON PLAN/OBJ: RO-C-02300/#11 & #12
REFERENCE: TRM 8.7.12, Explosive Gas Mixture, TRO 8.7.12

KA - 071000 K5.04
Waste Gas Disposal System (WGDS)
Knowledge of the operational implications of the following concepts as they apply to the Waste Gas Disposal System:
Relationship of hydrogen/oxygen concentrations to flammability
RO - 2.5 SRO - 3.1
CFR - 41.5 / 45.7
SCLR - 1F

KA Justification - Question tests knowledge of the operational (TRO) maximum concentrations of Hydrogen and Oxygen allowed in Waste Gas Decay tanks with are based on flammability limits.

61. 061 013/RO/OK/MODIFIED/NRC EXAM2006-025-13/002000 K6.02/3.6/3.8/H/3

Given the following conditions on Unit 1:

- The unit is at 20% power with a power ascension in progress.
- Rods are in Manual
- RCP 11 trips due to an overcurrent condition.
- No operator action has been taken.

Which ONE of the following describes the INITIAL reactor and the UNAFFECTED Loops response?

A. A reactor trip WILL NOT occur and the UNAFFECTED Loops Delta-T will lower.

B.✓ A reactor trip WILL NOT occur and the UNAFFECTED Loops Delta-T will rise.

C. A reactor trip WILL occur and the UNAFFECTED Loops Delta-T will lower.

D. A reactor trip WILL occur and the UNAFFECTED Loops Delta-T will rise.

ANSWER: B

A. INCORRECT. Delta-T will be higher.
B. CORRECT. A single RCP trip will not cause an automatic reactor trip below P-8. Once the RCP trips, the unaffected loops Delta-T rises to continue producing the same power level as before.
C. INCORRECT. No trip below 29% power and Delta-T will be higher.
D. INCORRECT. No trip below 29% power.

LESSON PLAN/OBJ: RO-C-TRANS4/#TRANS4A.2
REFERENCE: RO-C-TRANS4, RCS Loop Flow Transients

KA - 002000 K6.02
Reactor Coolant System (RCS)
Knowledge of the effect of a loss or malfunction of the following RCS components:
RCP
RO - 3.6 SRO - 3.8
CFR - 41.7 / 45.7
SCLR - 2RI

KA Justification - Question tests the knowledge that a loss of the RCP will have on the RCS temperature.

Modified Stem to Asks for UNAFFECTED loop response. Changed distractors to reflect unaffected loop response (Previous question asked about Loop 11). Correct answer Changed to B vs. A.

Question Source - Cook NRC Exam2006-025-13, INPO # 23437 Indian Point 3 (Unit) - 3/10/2003
Original Quest. KA - 002.k6.02

62. 062 062/RO/OK/DIRECT/NRC EXAM 2004-039-2/015000 A1.03/3.7/3.7/H/3

Unit 1 is conducting a reactor startup following a refueling outage.

Given the following plant conditions:

- Source Range and Intermediate Range Nuclear Instruments are slowly rising.
- Rods are in manual with no rod motion.
- Source Range Instruments read as follows:
  N-31: 2.1×10^4 cps.
  N-32: 2.0×10^4 cps.
- Intermediate Range Instruments read as follows:
  N-35: 2.5×10^{-11} amps.
  N-36: 1.0×10^{-9} amps.

Which ONE of the following best explains the indications?

A.✓ N-35 compensating voltage is set too high

B. N-35 compensating voltage is set too low

C. N-36 compensating voltage is set too high

D. N-36 compensating voltage is set too low

ANSWER: A

A. CORRECT. N-35 reads too low for the conditions given, compensating voltage is too high.
B. INCORRECT. N-35 reads too low.
C. INCORRECT. Overlap is proper for N-36.
D. INCORRECT. Overlap is proper for N-36.

LESSON PLAN/OBJ:RO-C-01300/#9
REFERENCE: RO-C-01300, HO-3 and Powerpoint TP-24

KA - 015000 A1.03
Nuclear Instrumentation System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the NIS controls including:
NIS power indication
RO - 3.7 SRO - 3.7
CFR - 41.5 / 45.5
SCLR - 2DR

KA Justification - Question tests knowledge of the expected (predict/monitor) NIS indications from Improper setting (control operation) of the discriminator voltage.

Question Source - NRC Exam 2004-039-2, AUDIT02-BOTH-16
Original KA - 015000 K6.02

63. 063 007/RO/OK/MODIFIED/RO24 AUDIT-058-7/017000 A2.02/3.6/4.1/H/3

The control room operators are performing 1-OHP-4023-FR-C.1, Inadequate Core Cooling . They are NOT able to establish high head ECCS flow.

Given the following plant conditions:

- SG depressurization proves to be ineffective.
- SG NR levels are stable at 30%.
- All core exit TCs are 1250°F and slowly rising.

The operators were attempting to establish conditions for RCP restart, but are unable to establish RCP seal injection or 200 psid across the #1 seal.

Which ONE of the following describes the current status of the fuel and the required actions?

A.✓ The fuel is NOT significantly damaged. The crew is required start one RCP at a time until core exit TCs are less than 1200°F.

B. The fuel is significantly damaged so the crew should NOT start the RCPs. They are required to open all PRZ PORVs and block valves.

C. The fuel is significantly damaged. The crew is required start all RCPs simultaneously to reduce core exit TCs to less than 1200°F.

D. The fuel is NOT significantly damaged so the crew should NOT start the RCPs. They are required to continue attempts to establish high head injection.

ANSWER: A

A. CORRECT. The 1200°F value was chosen because it was significantly below the point at which the fuel was damaged, but high enough that extreme measures are required to recover cooling. Since adequate SG levels for heat sink exist, the RCPs are started in an attempt to circulate coolant/steam through the SG tubes in an attempt to cool the RCS. Adequate support condition for the RCPs are desired but NOT required since Core Cooling is severely challenged.

B. INCORRECT. The fuel is not yet damaged. These actions are plausible since they are the RNO for step 22 if RCP or SG level is not available.

C. INCORRECT. The fuel is not yet damaged. Starting RCPs without all support conditions and at this temperature may damage RCPs so only 1 is started at a time. This is plausible since the loss of cooling is a severe challenge to the core and starting all RCPs may provide more cooling.

D. INCORRECT. This is plausible since the fuel is not damaged and the RCPs do not have all support conditions available (& opening the PRZ PORVs is a drastic step).

LESSON PLAN/OBJ: RO-C-EOP10/#6 & 9
REFERENCE: RO-C-EOP10, 01-OHP 4023.FR-C.1

KA - 017000 A2.02
In-Core Temperature Monitor (ITM) System
Ability to (a) predict the impacts of the following malfunctions or operations on the ITM System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Core damage
RO - 3.6 SRO - 4.1
CFR - 41.5 / 43.5 / 45.3 / 45.13
SCLR - 2RI

KA Justification - Question tests operator assessment of the core based on TC temperatures and knowledge of the required procedural actions.

Modified to add fuel damage aspect.

Question Source - RO24 Audit-058-7, RO23-108-3 (SRO95), Master bank 01ER1C7XX-14
Alt KA - EPE 074 EK3.11

64. 064 011/RO/OK/MODIFIED/NRC EXAM 2006-505-11/035000 A4.01/3.7/3.6/F/3

Given the following plant conditions on Unit 1:

- The unit is at 100% power and stable.
- Steam Generator Level Controls are in AUTOMATIC.
- Steam Generator #12 Feed Flow Channel 1, 1-FFC-220, is selected to the Steam Generator Level Control System.

If 1-FFC-220 **instantaneously** fails offscale low, which ONE of the following describes the expected plant response?

Note: Assume no operator action.

The Steam Generator Level Control system will:

A. initially lower feed flow and then slowly return SG#12 level to approximately program level.

B.✓ automatically transfer the SG#12 FW Regulating Valve Controller to Manual to maintain the current valve position.

C. initially raise feed flow and then slowly return SG#12 level to approximately program level.

D. automatically transfer the SG#12 FW Regulating Valve Controller to Feed Flow Channel 2, 1-FFC-221 to allow continued automatic operation.

ANSWER: B

A. INCORRECT. The FRV Controller will shift to manual. This is the response to a slow Steam Flow failure.
B. CORRECT. A sudden failure offscale will cause the Taylor Controller to shift to manual at the current position.
C. INCORRECT. The FRV Controller will shift to manual. This is the response to a slow FW Flow failure.
D. INCORRECT. The FRV Controller will shift to manual. This is the response to failures of instrumentation associated with the Digital Control Systems.

Note: Validated response on the simulator on 6-2-2008.

LESSON PLAN/OBJ: RO-C-05100 / #9, RO-C-AOP-D11/#RO-C-AOP0380412-E1
REFERENCE: RO-C-05100, Steam Generator System pg. 20-21, RO-C-ES07, Control Room Controllers and Recorders, RO-C-05100 TP75, 76, 78, 79, 82, 83

KA - 035000 A4.01
Steam Generator System (S/GS)
Ability to manually operate and/or monitor in the control room:
Shift of S/G controls between manual and automatic control, by bumpless transfer
RO - 3.7 SRO - 3.6
CFR - 41.7 / 45.5 to 45.8
SCLR - 1I

KA Justification - Question tests operator ability to monitor (identify expected actions) FW controller automatic transfer

Modified failure to be a FW Flow Transmitter and Changed from Blown fuse to Instantaneous failure.

Original Question - NRC EXAM 2006-505-11

65. 065 003/RO/OK/MODIFIED/NRC EXAM 2004-130-3/086000 2.4.8/3.8/4.5/F/3

Unit 2 was operating at 100% power when a fire occurred in the Unit 2 Charging Pump Area resulting in a loss of seal injection to all the RCPs.

The Shift Manager determines that a Reactor Trip is required based on the fire impacting control of the plant from the Control Room.

Which ONE of the following describes the correct Operator response?

Immediately trip the Reactor and RCPs and implement:

A.✓ 2-OHP-4023-E-0, Reactor Trip or Safety Injection.
12-OHP-4025-001-002, Fire Response Guidelines, for CCP area may be performed concurrently after the immediate actions are complete.

B. 2-OHP-4023-E-0, Reactor Trip or Safety Injection.
12-OHP-4025-001-002, Fire Response Guidelines, for CCP area is NOT needed since the EOP network addresses a loss of seal injection due to fire.

C. 2-OHP-4023-E-0, Reactor Trip or Safety Injection.
Steps from 12-OHP-4025-001-002, Fire Response Guidelines, for CCP area may NOT be performed until completion of 2-OHP-4023-ES-0.1, Reactor Trip Response.

D. 12-OHP-4025-001-002, Fire Response Guidelines, for CCP area until restoration of seal injection.
Perform 2-OHP-4023-E-0, Reactor Trip or Safety Injection, steps as time allows.

ANSWER: A

A. CORRECT. OHI-4023, Abnormal/Emergency Procedure User's Guide allows Abnormal Procedures to be implemented concurrently with Emergency Procedures.

B. INCORRECT. Performance of 02-OHP-4023-E-0 is required upon the reactor trip, but the operators may perform 12-OHP-4025-001-002, Fire Response Guideline, for CCP area to address fire in CCP area.

C. INCORRECT. User's Guide allows Abnormal Procedures to be implemented concurrently with Emergency Procedures.

D. INCORRECT. The Unit Supervisor should direct action of 02-OHP-4023-E-0.

LESSON PLAN/OBJ: RO-C-EOP01/#25, RO-C-AOP-D2/#RO-C-0660620801-E3
REFERENCE: OHI-4023 Abnormal/Emergency Procedure User's Guide pg. 10, 18, & 43, 12-OHP-4025-001-002, Fire Response Guidelines

KA - 086000 2.4.8
Fire Protection System (FPS)
Emergency Procedures/Plan
Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
RO - 3.8 SRO - 4.5
CFR - 41.10 / 43.5 / 45.13
SCLR - 1P

KA Justification - Question tests knowledge of how the Fire Response Guidelines (12-OHP-4025-001-002) prodedure is used in conjunction with the EOP/Rx Trip.

66. 066 021/RO/OK/DIRECT/RO25 AUDIT- 21/194001 2.1.42/2.5/3.4/H/3

Given the following plant conditions on Unit 2:

- The unit is in Mode 6 with refueling activities in progress.
- Containment purge is in service.
- A fuel element accidentally dropped into the cavity.
- All radiation monitor TRIP/BLOCK switches are in their NORMAL position.
- The Manipulator Crane area radiation monitor has a HIGH alarm.
- ERS-2305 and ERS-2405, Lower Containment Noble Gas Low Range Radiation Monitors, have a HIGH alarm.

Which ONE of the following actions would occur, assuming that operators follow the required actions of 12-OHP-4022-018-004, Irradiated Fuel Handling Accident In Containment Building - Control Room Actions and all equipment responds as designed?

A. Containment evacuation alarm sounds automatically.
Containment purge stops automatically.

B.✓ Containment evacuation alarm is manually actuated by the control room operator.
Containment purge stops automatically.

C. Containment evacuation alarm sounds automatically.
Containment purge is stopped manually by the control room operator.

D. Containment evacuation alarm is manually actuated by the control room operator.
Containment purge is stopped manually by the control room operator.

ANSWER: B

A. INCORRECT. Containment evacuation alarm is manually actuated by the control room operator.

B. CORRECT. OHP 4022.018.004, Step 1 directs the operator to actuate the Containment Evacuation alarm. For the protection of personnel, it is important to evacuate the affected area until radiation surveys can be completed. Step 2 follows up the alarm with a page announcement notifying all non-essential personnel to evacuate the containment. Steps 5 through 9 verify the containment purge and pressure relief systems are shutdown and isolated. This will limit the exposure of personnel outside containment.

C. INCORRECT. Containment evacuation alarm is manually actuated by the control room operator. Containment purge stops automatically.

D. INCORRECT. Containment purge stops automatically.

LESSON PLAN/OBJ: RO-C-1350\#4, RO-C-AOP-D9\#RO-C-AOP0630412-E3
REFERENCE: 12-OHP-4022-018-004, Irradiated Fuel Handling Accident In Containment Building - Control Room Actions, RO-C-AOP-D9, RO-C-1350 pg. 42-45

KA - 194001 2.1.42
Generic
Conduct of Operations
Knowledge of new and spent fuel movement procedures.
RO - 2.5 SRO - 3.4
CFR - 41.10 / 43.7 / 45.13
SCLR - 2RI

KA Justification - Question tests candidates knowledge of actions required per the procedure for an accident with irradiated fuel.

Question Source - RO25 AUDIT- 21, RO-22 AUDIT BOTH-76 (#68)
Original KA - APE036 G2.4.31, SYS 103 A2.04

67. 067 071/RO/OK/DIRECT/RO25 AUDIT-71/194001 2.1.43/4.1/4.3/H/3

Given the following plant conditions on Unit 1:

- The unit is operating at 100%.
- The feedwater temperature input to the LEFM thermal power calculation was incorrectly calibrated to 7ºF higher than actual feedwater temperature.
- Calibration of the power range nuclear instruments (NIs) is being performed.

How will LEFM power compare to actual thermal power and how will adjustment of the NIs be affected using the calculated value of LEFM?

A✓ Calculated thermal power is lower than actual power.
NI adjustment will result in indications that are less conservative.

B. Calculated thermal power is higher than actual power.
NI adjustment will result in indications that are less conservative.

C. Calculated thermal power is lower than actual power.
NI adjustment will result in indications that are more conservative.

D. Calculated thermal power is higher than actual power.
NI adjustment will result in indications that are more conservative.

ANSWER: A

A. CORRECT. Due to actual FW temperature being 7 degrees lower than indicated/calibrated FW temperature, calculated thermal power will indicate lower than actual thermal power. This will cause NI calibration to result in indications that are less conservative (i.e. farther from the trip setpoint).

B. INCORRECT. Calculated thermal power will be lower than actual thermal power.

C. INCORRECT. NI adjustment will be less conservative.

D. INCORRECT. Calculated thermal power will be lower than actual thermal power <u>AND</u> NI adjustment will be less conservative.

LESSON PLAN/OBJ: RO-C-GF19/#16
REFERENCE: GFE Thermodynamics Chapter 7, Heat Transfer (RO-C-GF19)

KA - 194001 2.1.43
Generic
Conduct of Operations
Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant system temperature, secondary plant, fuel depletion, etc.
RO - 4.1 SRO - 4.3
CFR - 41.10 / 43.6 / 45.6
SCLR - 3SPK

KA Justification - Question tests ability to determine effects on reactivity (& monitoring system) of differences in secondary plant temperatures based on knowledge of the calorimetric procedure.

Question Source - RO25 AUDIT-71, CATAWBA 2005
Original KA- Generic 2.2.34 2.8/3.2

68. 068 076/RO/OK/DIRECT/30725-PALISADES-2006/194001 2.2.13/4.1/4.3/F/2

Refer to the following list of valve operations:

1. Close discharge valve.
2. Close suction valve.
3. Open discharge valve.
4. Open suction valve.

Which ONE of the following describes the required sequence of valve operations when tagging out and subsequently restoring to service of a centrifugal pump?

	TAGOUT	**then**	**RESTORE**
A.	1,2	then	3,4
B.✓	1,2	then	4,3
C.	2,1	then	3,4
D.	2,1	then	4,3

ANSWER: B

A. INCORRECT. Opens discharge first.
B. CORRECT. Close discharge before suction, and open suction before discharge.
C. INCORRECT. Order for both tagout and restore wrong.
D. INCORRECT. Isolates suction first.

LESSON PLAN/OBJ: CP-C-0020/CP-C-0020-E5
REFERENCE: 12-OHP-2110-CPS-001, Clearance Permit System, CP-C-0020, Clearance Permit Writer

KA - 194001 2.2.13
Generic
Equipment Control
Knowledge of tagging and clearance procedures.
RO - 4.1 SRO - 4.3
CFR - 41.10 / 45.13
SCLR - 1P

KA Justification - Question tests knowledge of procedural clearance sequencing requirements.

Question Source - INPO Bank #30725 - PALSDES-2282006
Original Question KA - G 2.2.13

69. 069 036/RO/OK/NEW//194001 2.2.21/2.9/4.1/F/2

Who ensures that the scope of a post maintenance test is adequate to determine that equipment is operable following maintenance activity?

A. The Responsible System Engineer

B. The Work Week Manager

C. The Mechanical Maintenance Supervisor

D.✓ The Shift Manager or Work Control Center SRO

ANSWER: D

A. INCORRECT. System engineers are repsonsible for reviewing work on their systems, but do not deteremine what testing is required for operability.
B. INCORRECT. Work Week Manager only co-ordinate the activities that are required to be performed for post maintenece testing.
C. INCORRECT. Once returned to the correct position and being independently verified train B is considered operable - a test of the pump's auto start function is NOT required.
D. CORRECT. The SM or Work Control Center SRO both ensures that the scope of a post maintenance test is adequate to determine that equipment is operable following maintenance activity and also authorizes the performance of PMTs.

LESSON PLAN/OBJ: RO-C-ADM04/#2
REFERENCE: RO-C-ADM04, PMI-2294, Page 4 of 12

KA - 194001 2.2.21
Generic
Equipment Control
Knowledge of pre- and post-maintenance operability requirements.
RO - 2.9 SRO - 4.1
CFR - 41.10 / 43.2
SCLR - 1P

KA Justification - Question tests knowledge of post-maintenance testing responsibilities for equipment OPERABILITY.

70. 070 065/RO/OK/DIRECT/RO25 AUDIT-65/194001 2.2.2/4.6/4.1/H/3

Given the following plant conditions on Unit 2:

- A normal plant cooldown is in progress:
- RCS loop Tavg readings are all lowering with the current values:
  Loop 1: 538°F.
  Loop 2: 542°F.
  Loop 3: 537°F.
  Loop 4: 540°F.
- Steam header pressure is 900 psig and lowering.
- Steam Dump Mode Selector switch in STM PRESS MODE.
- Steam Dump Controller in MAN and set at 40% demand.
- ALL Steam Dumps are closed.

The operator momentarily places the Steam Dump Control Selector Train A and Control Selector Train B switches to BYPASS INTRLK and then releases them.

What is the expected status of the Steam Dump valves following the operator's actions?

A. All valves remain closed.

B.✓ The valves in group 1 are open and the valves in groups 2 and 3 are closed.

C. The valves in group 1 and 2 are open and the valves in group 3 are closed.

D. The valves in group 1 and 2 are open and the valves in group 3 are partially open.

ANSWER: B

A. INCORRECT. Group 1 valves are enabled below P-12 when Control Selector switches are place in Bypass Interlock position.
B. CORRECT. Below P-12 (2/4 Tavg less than 541 deg F) only the 3-Group 1 dump valves are available for service.
C. INCORRECT. Group 2 valves are not available less than P-12.
D. INCORRECT. Group 2 and 3 valves are not available less than P-12.

LESSON PLAN/OBJ: RO-C-NOP2 / #RO-C-NOP-E7, RO-C-05200/#6
REFERENCE: SOD-05200-001

KA - 194001 2.2.2
Generic
Equipment Control
Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.
RO - 4.6 SRO - 4.1
CFR - 41.6 / 41.7 / 45.2
SCLR - 2RI

KA Justification - Question tests candidates ability to determine status of steam dump system response based on plant conditions and control manipulations.

Question Source - RO25 AUDIT-65, MAST 01052C0012
Original KA- 041000 2.1.23 3.9/4.0 CFR 45.2/45.6

71. 071 001/RO/OK/DIRECT/NRC EXAM 2002-118-1/194001 2.3.11/3.8/4.3/F/2

The control room operators are responding to a Steam Generator Tube Rupture. In order to cool down the RCS and establish required subcooling margin, the operators dump steam to the condenser using the intact SGs.

Which ONE of the following describes why this method of RCS cooldown is preferred over dumping steam through the PORVs of the intact SGs?

A.✓ Minimizes radiological releases.

B. Minimizes thermal shock to the reactor vessel.

C. Minimizes shrink experienced by the RCS.

D. Minimizes RCS subcooling requirements.

ANSWER: A

A. CORRECT. Dumping steam to the condenser will keep the radioactivity contained within the plant. Prior to the ruptured SG isolation, the leaking RCS would have mixed with the secondary and raised the activity levels of the intact SGs.
B. INCORRECT. The magnitude of the cooldown will be the same with both the SG PORVs and the condenser. Using the condenser may actually cooldown the RCS faster.
C. INCORRECT. The magnitude of the cooldown will be the same with both the SG PORVs and the condenser. This is directly related to the amount of RCS shrink.
D. INCORRECT. RCS Subcooling requirements are the same.

Note: Distracters are plausible since the SG PORVs may provide greater cooling at maximum rate and are typically less controllable than the fine control provided by the steam dumps.

LESSON PLAN/OBJ: RO-C-EOP08 / #12, 19
REFERENCE: RO-C-EOP08, 12-OHP-4023-E-3, Steam Generator Tube Rupture Background

KA - 194001 2.3.11
Generic
Radiation Control
Ability to control radiation releases.
RO - 3.8 SRO - 4.3
CFR - 41.11 / 43.4 / 45.10
SCLR - 1F

KA Justification - Question examines the control of release during a SGTR by using the steam dumps as the preferred method of cooldown.

Question Source - Cook RO24 Audit-073-11, NRC Exam 2002-118-1 (RO#95), INPO Bank #19590 - COOK-5212001

72. 072 004/RO/OK/MODIFIED/NRC EXAM 2004-100-3/194001 2.3.15/2.9/3.1/H/3

Given the following plant conditions on Unit 1:

- Containment Purge System is operating in the CLEAN-UP MODE.
- A power supply spike caused the HIGH alarm on VRS-1201, Upper Containment Normal Range Area Monitor to actuate.

Which ONE of the following describes the response of the Containment Ventilation System to this alarm?

A. Containment ventilation isolation valves 1-VCR-101 through 1-VCR-107 close
1-HV-CIPS-1, Containment Instrument Room Purge Supply Fan, trips

B. Containment ventilation isolation valves 1-VCR-101 through 1-VCR-107 close
1-HV-CPS-1/2, Containment Purge Supply Fans 1 and 2, trip
1-HV-CPX-1/2, Containment Purge Exhaust Fans 1 and 2, trip
1-HV-CPR-1, Containment Pressure Relief Fan, trips
1-HV-CIPS-1, Containment Instrument Room Purge Supply Fan, trips

C. Containment ventilation isolation valves 1-VCR-201 through 1-VCR-207 close
1-HV-CIPS-1, Containment Instrument Room Purge Supply Fan, trips

D.✓ Containment ventilation isolation valves 1-VCR-201 through 1-VCR-207 close
1-HV-CPS-1/2, Containment Purge Supply Fans 1 and 2, trip
1-HV-CPX-1/2, Containment Purge Exhaust Fans 1 and 2, trip
1-HV-CPR-1, Containment Pressure Relief Fan, trips
1-HV-CIPX-1, Containment Instrument Room Purge Exhaust Fan, trips

ANSWER: D

A. INCORRECT. These are the actions from the VRS-1101 monitor actuation.
B. INCORRECT. The Purge supply and exhaust and pressure relief fan will NOT trip
C. INCORRECT. The Outside containment isolation valves will not close
D. CORRECT. While Operating in the Clean-UP mode the Radiation Monitor switches are unblocked allowing actuations. VRS-1201 closes the Outside Containment Isolation valves and trips the listed fans.

LESSON PLAN/OBJ: RO-C-02800/#9, RO-C-01350/#4
REFERENCE: 12-OHP-4024-139 Annunciator #139 Response: Radiation Drop 1 & Drop 2, RO-C-01350

KA - 194001 2.3.15
Generic
Radiation Control
Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
RO - 2.9 SRO - 3.1
CFR - 41.12 / 43.4 / 45.9
SCLR - 3PEO

KA Justification - Question requires candidate to know the failure mode of the fixed area monitor and the automatic actions associated with a failure alarm.

Question Source - NRC EXAM 2004-100-3, AUDIT02-BOTH31

Modified from VRS-1101 to VRS-1201. This changes the correct answer from A to D, whihc are different valves and different fans that trip from this radiation monitor.

73. 073 001/RO/OK/DIRECT/NRC EXAM 2002-124-1/194001 2.4.45/4.1/4.3/F/2

Given the following plant conditions on Unit 1:

- The unit has experienced a loss of offsite power.
- You are an extra operator and are assigned to address the annunciators on Panel 119, Station Auxiliary AB.

Which ONE of the following would you address FIRST based on the OHI-4000 Annunciator Priority system?

The Annunciator with:

A. a Red "C" on the lens.

B.✓ a Red lens with a Purple slash.

C. a White lens with a slash in the lower right corner.

D. an Orange dot on the lens.

ANSWER: B

A. INCORRECT. These alarms mean that compensatory actions may be required.
B. CORRECT. The red lens is a first priority annunciator. Adding the purple slash means a condition that would not allow the EDG to start or deliver power.
C. INCORRECT. These are seal-in alarms.
D. INCORRECT. An Orange dot indicates a second priority lens.

LESSON PLAN/OBJ: RO-C-ADM02 / #25
REFERENCE: OHI-4000 Attachment 1 Alarm Response Section 3.9

KA - 194001 2.4.45
Generic
Emergency Procedures/Plan
Ability to prioritize and interpret the significance of each annunciator or alarm.
RO - 4.1 SRO - 4.3
CFR - 41.10 / 43.5 / 45.3 / 45.12
SCLR - 1T

KA Justification - Question tests the ability to determine which alarm has the highest priority based on the coding used at the station.

Question Source - NRC Exam 2002-124-1 (Q#98)

74. 074 059/RO/OK/DIRECT/30419-KEWAUNEE-2006/194001 2.4.50/4.2/4.0/F/3

Given the following plant conditions:

- Plant is at 75% power.
- RCS boron concentration is 480 ppm.
- Control Bank D is at 200 steps.
- Control Rod Bank Selector is in AUTO.
- Turbine load is being raised by 200 MWe.
- Ann. 210, Drop 26, ROD CONTROL URGENT FAILURE is LIT.
- Rod motion stops.
- The AEO reports the URGENT FAILURE light on the 1BD Power Cabinet is LIT.

What action is taken for this condition?

A. Continue the power escalation using dilution.

B. Manually trip the reactor and go to 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

C✓ Stop the power escalation and maintain Tave using Turbine load reduction and/or adjusting RCS Boron Concentration.

D. Place rods in MANUAL and reduce reactor power below 50% within 4 hours using Turbine Load Reduction and boration.

ANSWER: C

A. INCORRECT. This action is capable of being performed but is NOT allowed in accordance with reactivity management guidelines and the procedure.

B. INCORRECT. This is the immediate action required per the AOP if rods are moving and fail to stop when the Bank Selector is taken to MAN.

C. CORRECT. The power escalation should be stopped to investigate the plant conditions. OHP-4022-012-001, Failure of A Control Bank to Move Directs using a Turbine load reduction or Boron Adjustment to stabilize the plant.

D. INCORRECT. With this failure, rod motion IN or OUT in MAN or AUTO is prohibited by the system. If the rods indicated a misalignment, the appropriate action would be to realign the rods or determine the core power peaking factors within four hours. If this can NOT be done within the given time frame, then reactor power must be reduced to < 50%.

LESSON PLAN/OBJ: RO-C-AOP-D11/RO-C-0120210412-E2
REFERENCE: 2-OHP-4024-210 Drop 26, 2-OHP-4022-012-001, Failure of A Control Bank to Move

KA - 194001 2.4.50
Generic
Emergency Procedures/Plan
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.
RO - 4.2 SRO - 4.0
CFR - 41.10 / 43.5 / 45.3
SCLR - 1P

KA Justification - Question requires candidate to identify actions required based on alarm response and procedural requirements.

Question Source - INPO Bank #30419 - KEWAUNEE-222006
Original Question KA - 001 2.4.50

75. 075 012/RO/OK/MODIFIED/NRC EXAM 2004-109-2/194001 2.4.46/4.2/4.2/H/3

A reactor trip with a safety injection occurred due to a feed line break on SG#22. The crew is performing actions of 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

Steam Generator Aux Feedwater Flows were indicating as follows:

	SG21	SG22	SG23	SG24
Flow Instrument	FFI-210	FFI-220	FFI-230	FFI-240
Flow in PPH	200×10^3	Pegged High	200×10^3	200×10^3

Ann. 214, Drop 9, TDAFP DISCHARGE FLOW HIGH, has just alarmed.

Which ONE of the following responses is correct given these conditions?

A. Do NOT trip TDAFP. Manually throttle AFW Flow to ALL 4 SGs until the Discharge Flow High Alarm clears. This alarm indicates that Aux Feed Flow Retention has FAILED.

B.✓ Do NOT trip TDAFP. Verify that AFW Flow to ALL 4 SGs has automatically throttled as expected for Aux Feed Flow Retention.This alarm indicates that Aux Feed Flow Retention has ACTUATED.

C. Trip TDAFP. This alarm indicates that the feed line break is on the Aux Feed Line.

D. Trip TDAFP. This alarm indicates that Aux Feed Flow Retention has FAILED.

ANSWER: B

A. INCORRECT. The Alarm Indicates that Flow Retention has Actuated and will Automatically Throttle the valves.

B. CORRECT. Upon High AFP flow to a SG (>975 gpm) the flow retention circuit will throttle the AFP valves closed to prevent pump runout. This is an expected alarm given these conditions. The pump should continue to operate after verifying that flow retention is properly operating.

C. INCORRECT. This alarm is expected for this condition. The AFP should NOT be stopped.

D. INCORRECT. The alarm indicates that flow retention has actuated. The AFP should NOT be stopped.

LESSON PLAN/OBJ: RO-C-05600/#12
REFERENCE: 02-OHP-4024.214, Annunciator #214 Drop 9 TDAFP Discharge Flow High; RO-C-05600, Auxiliary Feedwater System pg. 33-35

KA - 194001 2.4.46
Generic
Emergency Procedures/Plan
Ability to verify that the alarms are consistent with the plant conditions.
RO - 4.2 SRO - 4.2
CFR - 41.10 / 43.5 / 45.3 / 45.12
SCLR - 3PEO

Modified to Change to TDAFP from East MDAFP. Changed distractor A & and Answer B to relect alarm failed/actuated and Manual/Auto Throttling.

KA Justification - Question requires Candidate to determine if the AFW system is responding appropriately based on alarms and indications and to identify the correct action.

Question Source - NRC EXAM 2004-109-2 modified slightly to remove SRO decision.

76. 076 121/SRO/OK/DIRECT/SALEM 2002 - 23151/000025 AA2.04/3.3/3.6/H/3

Given the following plant conditions on Unit 2:

- Unit is in MODE 4 cooling down on RHR.
- RCS Temperature - 340°F.
- RCS pressure - 300 psig lowering.
- Pressurizer level - 22% lowering.
- Containment pressure - 0.1 psig.
- VRS-2505, U2 Vent Noble Gas is in Alarm.
- VRS-2503, U2 Vent Iodine Monitor radiation levels are trending higher.
- SG levels stable at - 42% (21); 40% (22); 43% (23); 40% (24).

Which ONE of the following identifies the problem and the associated action?

A.✓ A LOCA has occurred on the suction of the RHR pump. Enter 2-OHP-4022-002-015, Mode 4 LOCA.

B. LTOP (Low Temperature Over Pressure) actuated and one PORV is stuck open. Enter 2-OHP-4022-002-015, Mode 4 LOCA.

C. Letdown line pressure relief valve has failed open. Enter 2-OHP-4022-002-020, Excessive Reactor Coolant Leakage

D. A LOCA has occurred in the area of the Regenerative Heat Exchanger. Enter 2-OHP-4022-002-020, Excessive Reactor Coolant Leakage

ANSWER: A

A. CORRECT. During any LOCA, RCS pressure and inventory will fall. Rising indication on the Aux Bldg. Radiation monitors is indicative of the LOCA outside containment (RHR pump suction).

B. INCORRECT. When LTOPs operation results in a PORV opening, RCS pressure will drop, but PZR level should rise due to voiding in the reactor vessel head. Initial conditions do NOT support LTOP auto operation.

C. INCORRECT. Failure of IRV-300 open results in increased diversion of RHR flow (RCS inventory) to letdown but the flow is contained and would not lead to rising Aux Bldg. RMS indications.

D. INCORRECT. This leak is inside containment.

LESSON PLAN/OBJ: RO-C-AOP-D16/RO-C-AOP0250412-E2
REFERENCE: 02-OHP-4022-002-015, Mode 4 LOCA, RO-C-AOP-D16

KA - 000025 AA2.04
Loss of Residual Heat Removal System (RHRS)
Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System:
Location and isolability of leaks
RO - 3.3 SRO - 3.6
CFR - 41.7 / 41.10 / 43.5 / 45.13
SCLR - 2RI

KA Justification - Question requires candidate to determine the location of RHR leak and selection of appropriate procedure for isolation (SRO).

Question Source - INPO Bank #23151 - SALEM-1142002
Original Question KA - ..025.AA2.04

77. 077 008/SRO/OK/NEW//000026 AA2.02/2.9/3.6/H/3

Unit 2 was operating at 100% power when indications of a lowering CCW Surge tank required entry into 2-OHP-4022-016-001, Malfunction of the CCW System.

The Crew has started the West CCW pump, split the East and West Headers aligning the Miscellaneous Services Header to the East Header.

The following Surge Tank Level Recorder conditions exist:

	CLR-410	**CLR-411**
Reading	18"	48"
Trend	Lowering	Stable

An AEO reports that a CCW leak of approximately 150 gpm has been identified near the South Spent Fuel Pool Heat Exchanger.

Which ONE of the following describes the leak location and the required actions?

The leak is located on the:

A. Miscellaneous Services Header. Trip the Reactor, Stop both CCW Pumps, and Implement 2-OHP-4022-016-004, Loss of CCW along with 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

B. East Safeguards Header. Shutdown the East CCW pump and align the Miscellaneous Services Header to the West Safeguards Header.

C. West Safeguards Header. Shutdown the West CCW pump and the equipment cooled by the West Header.

D.✓ Miscellaneous Services Header. Trip the Reactor, Trip the RCPs, and isolate the Miscellaneous Services Header while performing 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

ANSWER: D

A. INCORRECT. During the initial train split, the Misc. Header is aligned to the East Safeguards Header. The West CCW pump is isolated from the Leak and does NOT need to be tripped.
B. INCORRECT. The SFP cooling is supplied from the Misc. Header.
C. INCORRECT. The SFP cooling is supplied from the Misc. Header.
D. CORRECT. During the initial train split, the Misc. Header is aligned to the East Safeguards Header. The SFP cooling is supplied from the Misc. Header. Based on the Leak rate isolation of the Misc. Header requires that the Reactor be tripped and the RCPs stopped since they loose cooling.

LESSON PLAN/OBJ: RO-C-AOP-D8\RO-C-AOP0420412-E3, RO-C-01600\#2
REFERENCE: 02-OHP-4022-016-001, Malfunction of the CCW System,
SOD-01600-001, RO-C-01600, RO-C-AOP-D8

KA - 000026 AA2.02
Loss of Component Cooling Water (CCW)
Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water:
The cause of possible CCW loss
RO - 2.9 SRO - 3.6
CFR - 41.7 / 41.10 / 43.5 / 45.13
SCLR - 3SPR

KA Justification - Question requires Candidate to determine the leak location and identify the equipment lost and actions required.

78. 078 007/SRO/OK/MODIFIED/NRC EXAM 2007-74/000040 AA2.03/4.6/4.7/H/3

Given the following plant conditions on Unit 2:

- Reactor power: 62% and rising.
- RCS pressure: 2220 PSIG and lowering.
- Auctioneered High Tavg: 560°F and lowering.
- Turbine power: 520 MWe and lowering.

Based on the above plant indications, what event is occurring and what are the required actions/procedures to address the event?

A.✓ A Steamline Break requires a reactor trip and 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

B. A Steamline Leak requires a rapid power reduction per 2-OHP-4022-001-006, Rapid Power Reduction Response.

C. A Small Break RCS LOCA requires a reactor trip and 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

D. An RCS Leak requires implementation of 2-OHP-4022-002-020, Excessive Reactor Coolant Leakage.

ANSWER: A

A. CORRECT. Reactor power is rising, indicating positive reactivity event. Electric load is lowering, indicating loss of steam to the turbine. Turbine power should be closer to 700 Mw with Tavg closer to 563°F based on this reactor power. Based on this degree of mismatch a reactor trip is required (due to the size of the steam break).

B. INCORRECT. A Steam Leak is occuring, but based on the large mismatch a reactor trip is required.

C. INCORRECT. Plausible since RCS pressure is lowering, but Reactor power is rising - indicating positive reactivity event.

D. INCORRECT. Plausible since RCS pressure is lowering, but Reactor power is rising - indicating positive reactivity event.

LESSON PLAN/OBJ: RO-C-EOP07/#4
REFERENCE: RO-C-EOP07, Secondary Side Breaks E-2 series EOPs & Background Information pg. 11-12, OHI-4023 pg. 22, TDB 2-Figure 2.4

KA - 000040 AA2.03
Steam Line Rupture
Ability to determine and interpret the following as they apply to the Steam Line Rupture: Difference between steam line rupture and LOCA
RO - 4.6 SRO - 4.7
CFR - 41.7 / 41.10 / 43.5 / 45.13
SCLR - 2DR

KA Justification - Question requires candidate to differentiate between a Steam Line Break & LOCA. Identification of the magnitude and procedural selection apply to SROs Only.

Modified Stem by changing parameters and requiring procedural selection. Changed distractors B & D to make leaks (smaller magnitude of A & C) and all distractors to include procedures.

Question Source - Cook NRC Exam 2007-74, Audit RO24-012-07

79. 079 095/SRO/OK/NEW/NEW/000029 2.4.22/3.6/4.0/F/2

Which one of the following sets of critical safety functions (CSFs) contain ONLY functions which form the bases for protection of the Fuel Matrix/Cladding Barrier and are in the correct order of priority?

A. 1. Core Cooling 2. RCS Inventory 3. Heat Sink

B. 1. Subcriticality 2. Heat Sink 3. RCS Integrity

C.✓ 1. Subcriticality 2. Core Cooling 3. Heat Sink

D. 1. Heat Sink 2. RCS Integrity 3. RCS Inventory

ANSWER: C

A. INCORRECT. These functions are all correct but are in the wrong order.
B. INCORRECT. Wrong functions. An operator may think that the cladding is jeopardized by a loss of RCS Integrity (excessive heating due to loss of cooling medium).
C. CORRECT. Subcriticality, Core Cooling, and Heat sink (along with Inventory) are the associated with the Fuel Cald matrix and are in the correct order.
D. INCORRECT. Wrong functions. An operator may think that the cladding is jeopardized by a loss of RCS Integrity and RCS Inventory(excessive heating due to loss of cooling medium).

Lesson Plan\Obj: RO-C-EOP01/ #18 &19

References: RO-C-EOP01 TP 54 & 55, ERG Executive Vol. Description Pg. 9-11

KA- 000029 2.4.22
Anticipated Transient Without Scram (ATWS)
Emergency Procedures/Plan
Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.
RO - 3.6 SRO - 4.4
CFR - 41.7 / 41.10 / 43.5 / 45.12
SCLR - 2RI

KA Justification - Question tests knowledge of the bases used for the safety function (strategies & procedures used to address challenges to the cladding matrix - SRO Question)

80. 080 015/SRO/OK/NEW//000057 2.4.46/4.2/4.2/H/4

Unit 2 was at 60% power when the following conditions are noted:

- Ann. 219, Drop 29, CRID 3 INVERTER ABNORMAL is alarming.
- Ann. 219, Drop 30, CRID 4 INVERTER ABNORMAL is alarming.
- SG Feed Flows are Stable.

A scan of the NIS drawers indicates the following:

- Source Range N-31 and N-32 at zero.
- Source Range N-21 and N-23 near full scale.
- Intermediate Range N-35 and N-36 near full scale.
- Power Range N-41, N-42 and N-44 at mid-scale.
- Power Range N-43 at zero.

Which ONE of the following describes this failure and what actions/procedures are required?

A. CRID 3 & 4 have transferred to the Alternate Power Supply. Battery 2AB has failed, enter 2-OHP-4022-082-002A, Loss of Power to 250VDC Bus 2AB ONLY.

B.✓ CRID 4 has transferred to the Alternate Power Supply. CRID 3 has Failed to Transfer. Refer to 2-OHP-4021-082-008, Operation of CRID Power Supplies.

C. CRID 3 has transferred to the Alternate Power Supply. CRID 4 has Failed to Transfer. Refer to 2-OHP-4021-082-008, Operation of CRID Power Supplies.

D. CRID 3 & 4 have failed to transfer to the Alternate Power Supply. Initiate a Reactor Trip and enter 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

ANSWER: B

A. INCORRECT. The Failure of N-43 indicates that CRID 3 is lost. N-23 is powered from Unit 1. A loss of Battery 2AB would have caused a FW Isolation.

B. CORRECT. The Failure of N-43 indicates that CRID 3 is lost. N-23 is powered from Unit 1.

C. INCORRECT. The Failure of N-43 indicates that CRID 3 is lost. N-23 is powered from Unit 1. A Loss of CRID 4 would cause a Reactor Trip due to FW Isolation.

D. INCORRECT. The Failure of N-43 indicates that CRID 3 is lost. N-23 is powered from Unit 1. A Loss of CRID 4 would cause a Reactor Trip due to FW Isolation.

LESSON PLAN/OBJ: RO-C-AOP-D13/#RO-C-0820360101-T1
REFERENCE: 02-OHP-4024-219 Drops 29 & 30, RO-C-AOP-D13,
02-OHP-4021-082-008, Operation of CRID Power Supplies

KA - 000057 2.4.46
Loss of Vital AC Electrical Instrument Bus
Emergency Procedures/Plan
Ability to verify that the alarms are consistent with the plant conditions.
RO - 4.2 SRO - 4.2
CFR - 41.10 / 43.5 / 45.3 / 45.12
SCLR - 2RI

KA Justification - Question presents the candidate with alarms & failures and requires candidate to determine what the failure was (verify alarms consistent with conditions = Identify conditions which caused alarms). Procedural selection/actions included for SRO criteria.

81. 081 014/SRO/OK/NEW/NEW/000007 2.2.38/3.6/4.5/H/4

Given the following on Unit 2:

- You are the Unit Supervisor
- The Turbine driven AFW is INOPERABLE due to a failed trip/throttle valve.
- It was discovered that the West Motor Driven AFW Pumps bearing was failed rendering it INOPERABLE.
- TS 3.7.5 Auxiliary Feedwater was entered due to 2 AFW trains INOPERABLE.

Power was being reduced to 15% in preparation for a Plant Shutdown as required per TS 3.7.5 Auxiliary Feedwater (AFW) System when the reactor was tripped.

The plant has been stabilized with the East Motor Driven Driven AFW pump operating and the crew has completed 02-OHP-4023-ES-0.1.

- The STA states that you should stabilize the plant at 375°F and NOT enter Mode 4.

Do you agree or disagree <u>and</u> why?

A. Agree, Tech Spec 3.0.4 prohibits mode changes if all applicable tech specs for that mode are not met.

B. Agree, with 2 trains of AFW INOPERABLE the plant should be maintained in the current mode until at least 2 AFW pumps are OPERABLE.

C. Disagree, a standing Notice of Enforcement Discretion is in place to allow the plants to continue to lower modes even if they don't meet all Tech Specs.

D✓ Disagree, the plant should be placed into Mode 4 where only 1 AFW pump is required to be OPERABLE.

ANSWER: D

A. INCORRECT. Tech Spec 3.0.4 allows you to pass through lower modes without meeting all conditions ONLY if you are complying with a required action statement of another Tech Spec. Plausible since TS does not allow you to enter higher modes.

B. INCORRECT. The TS 3.7.5 contains a note that 3.0.3 and all LCOs requiring Mode changes are suspended if ALL 3 AFW trains are Inoperable. There are only 2 trains Inoperable.

C. INCORRECT. A standing Notice of Enforcement Discretion does NOT exist. Plausible since a NED may be issued as a short term solution for failing to meet TS with adequate justification.

D. CORRECT. Tech Spec 3.0.4 allows you to pass through lower modes without meeting all conditions ONLY if you are complying with a required action statement of another Tech Spec. Plausible since TS does not allow you to enter higher modes.

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Lesson Plan/Objective:RO-C-ADM01 / #13 & 19
Reference: Tech Spec LCO 3.7.5 & 3.0.4 and Bases B 3.0.4

KA - 000007 2.2.38
Reactor Trip - Stabilization
Equipment Control
Knowledge of conditions and limitations in the facility license.
RO - 3.6 SRO - 4.5
CFR - 41.7 / 43.1 / 45.13
SCLR - 3SPK

K/A Justification - Requires the knowledge of the facility license (LCO) following a reactor trip with AFW Inoperability.

Similar to NRC Exam 2004-123, 2002-106 except with AFW instead of RHR & RCS loops.

82. 082 002/SRO/OK/NEW//000068 AA2.11/4.3/4.4/H/3

The plant was at 100% power when a Control Room fire caused a Loss of Offsite Power and forced the evacuation of the Control Room. A cooldown at 14°F/hr is in progress. The operator has been dispatched to initiate Process Monitoring from the LSI Panels using 1-OHP-4025-LS-1-1. Startup LSI Panels and Process Monitoring.

The current conditions exist:

- RCS T_{hot} is 400°F.
- SG Pressures are 150 psig.
- RCS pressure is 1700 psig.

Which ONE of the following actions should the Local SRO direct the local operator at SG LSI panels to perform and why?

Note: 1-OHP-4025-LS-1-1, Startup LSI Panels and Process Monitoring is attached.

A. No action is required since the 50°F subcooling margin is satisfied.

B. Cooldown the RCS to attempt to reestablish Natural Circulation.

C. Depressurize RCS to minimize excess subcooling above the 50°F requirement.

D✓ Cool down RCS T_{hot} to establish a minimum of 220°F subcooling margin.

ANSWER: D

A. INCORRECT. The required subcooling margin is 220°F when on natural circulation. Subcooling is based on RCS pressure and Hot Leg temps.
B. INCORRECT. Natural circulation is occurring since the Tcold is 365 psig.
C. INCORRECT. The required subcooling margin is 220°F when on natural circulation.
D. CORRECT. The required subcooling margin is 220°F when on natural circulation. Subcooling is based on RCS pressure and Hot Leg temps.

Tsat 1715 psia = 614°F

LESSON PLAN/OBJ: RO-C-EC02/#4
REFERENCE: 1-OHP-4025-LS-1-1

Attachment Provided: 01-OHP-4025-LS-1-1. Startup LSI Panels and Process Monitoring

KA - 000068 AA2.11
Control Room Evacuation
Ability to determine and interpret the following as they apply to the Control Room Evacuation:
Indications of natural circulation
RO - 4.3 SRO - 4.4
CFR - 41.7 / 41.10 / 43.5 / 45.13
SCLR - 3SPR

KA Justification - Requires the ability determine if adequate subcooling exists to support natural circulation during a fire causing evacuation of the control room.

83. 083 027/SRO/OK/DIRECT/RO25AUD-27/00WE07 EA2.2/3.3/3.9/H/3

Conditions at 1400 hrs:

- Unit 2 safety injection occurred due to a LOCA
- Reactor coolant system pressure is 915 psig
- RCS T-Cold temperatures are 535°F
- Core Exit Thermocouples (CETC) are 565°F
- Crew has implemented 02-OHP-4023-FR-C.2, Response to Degraded Core Cooling
- The SM has directed a cooldown to allow RHR to be placed in service

Assuming temperatures have been stable since 1300 hrs, what is the maximum cooldown rate allowed per 02-OHP-4023-FR-C.2, and what is the lowest temperature that could be attained by 1600 hrs using that rate?

A. Cooldown at 60°F/hr to 415°F on RCS T-Colds

B. Cooldown at 60°F/hr to 445°F on CETCs

C.✓ Cooldown at 100°F/hr to 335°F on RCS T-Colds

D. Cooldown at 100°F/hr to 365°F on CETCs

ANSWER: C

A. INCORRECT. 60°F/hr Cooldown Rate - 60°F/hr is plausible because if the normal cooldown limit in NOPs.
B. INCORRECT. 60°F/hr & CETC temps used. The reference temperature to use is the T-Cold, NOT Core exit temperature.
C. CORRECT. This procedure allows 100°F/hr. (TS 3.4.3 limit). The reference temperature to use is the T-Cold
D. INCORRECT. CETC temps used. The reference temperature to use is the T-Cold, NOT Core exit temperature.

LESSON PLAN/OBJ: RO-C-EOP10\#20
REFERENCE: OHP-4023-FR-C.2 Step 14 and Background, RO-C-EOP10 pg. 80

Critical Task FR-C-2—A: Depressurize the SGs to atmospheric pressure while maintaining the cooldown rate in the RCS cold legs at less than 100°F/hr in order to initiate accumulator injection and to establish RHR injection to the RCS before an extreme (RED path) challenge develops to the Core Cooling CSF.

KA - 00WE07 EA2.2
Saturated Core Cooling
Ability to determine and interpret the following as they apply to the Saturated Core Cooling:
Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments
RO - 3.3 SRO - 3.9
CFR - 41.7 / 41.10 / 43.5 / 45.13
SCLR - 3SPK
KA Justification - Question tests knowledge of Procedural/TS limits and the indications used to adhere to those limits.

Question Source - RO25 Audit -27, CATAWBA2005

84. 084 007/SRO/OK/DIRECT/RO24 AUDIT-091-7/000003 2.4.50/4.2/4.0/H/3

A reactor Startup is in progress on Unit 1. The crew has just completed recording critical data. When the RO begins to withdraw control rods to raise reactor power, the IR NIS indication suddenly drops by 1/3 decade and continues to decrease at a negative (-).3 DPM.

The following conditions exist:

- There is no significant change in RCS Tave.
- The Control Bank D step counters now read 131 steps for both D1 and D2 groups.
- IRPI indicators for Control Bank D1 Rods D-4, D-12, M-4, and M-12 indicate 0 steps.

Which ONE of the following has occurred based on these indications and what are the required actions?

A. The control bank step counters and associated IRPI indicators, along with the NIS indications are consistent with multiple dropped rods.
Direct the RO to Manually re-insert all control banks and enter 01-OHP-4022-012-005, Dropped or Misaligned Rod.

B.✓ The control bank step counters and associated IRPI indicators, along with the NIS indications are consistent with multiple dropped rods.
Direct the RO to perform a manual reactor trip and enter 01-OHP-4023-E-0, Reactor Trip or Safety Injection.

C. An ATWS condition has occured since more than a single dropped rod would have resulted in a reactor trip.
Direct the RO to perform a manual reactor trip and enter 01-OHP-4023-S-1, Response to Nuclear Power Generation/ATWS.

D. The control bank D group step counter has failed, it should also read 0 steps if the rods in this group are fully inserted.
Direct the RO to stop rod withdrawal and enter 01-OHP-4022-012-005, Dropped or Misaligned Rod.

ANSWER: B
The IPRI indications and the lowering NIS indicates that multiple rods have dropped. The reactor did not trip automatically since <5% PR change. The operator will need to trip manually. (Multiple Rod drops require a Reactor Trip)

A - Incorrect - Multiple Rod drops require a Reactor Trip per 01-OHP-4022-012-005, Dropped or Misaligned Rod. Plausible since at low power the reactor will not auto trip & the indications are correct.
C - Incorrect - The reactor did not trip automatically because power is too low to receive a negative rate trip. (<5% PR change). The operator will need to trip manually. Plausible since this is generally true at higher power levels.
D - Incorrect - The Group step counter indicates demand position. Plausible since no RCS Tave changes were noted and the rods are fully inserted.
Lesson Plan/Objective:RO-C-01200/#23, RO-C-AOP-D8/RO-C-AOP-0240412-E2
Reference:RO-C-AOP-D8, Abnormal Operating Procedures – Day 8 TP-14, 01-4022-012-005 Step 1

KA - Dropped Control Rod
Emergency Procedures/Plan
Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.
RO - 4.2 SRO - 4.0
CFR - 41.10 / 43.5 / 45.3
SCLR - 3SPR

KA Justification - Question tests ability to determine if which alarm is valid & determine actions required.

Original Question Source: INPO-DIRECT 21426-BWD02, Modified NRC04-038-2 (#38), RO24 AUDIT-091-7
Original KA - 014000 K1.02

85. 085 002/SRO/OK/DIRECT/AUDIT RO23 Q#86/000061 2.4.4/4.5/4.7/H/3

Chemistry had just confirmed two leaking fuel rods on Unit 1 when a LOCA occurred.

Given the following plant conditions:

- 1-OHP-4023-E-0 Reactor Trip or Safety Injection is complete.
- 1-OHP-4023-E-1 Loss of Reactor or Secondary Coolant is complete.
- 1-OHP-4023-ES-1.2 Post LOCA Cooldown and Depressurization is in effect.
- All Red and Orange Paths have been addressed.
- Lower Containment high range area monitors, (VRA-1310/1410) are reading 10R/hr.
- Pressurizer level is 0%.

The SM directs you to implement the highest priority Yellow Path procedure.

Note:
- 1-OHP-4023-FR-I.2, Response to Low Pressurizer Level
- 1-OHP-4023-FR-Z.3, Response to High Containment Radiation Level

Which ONE of the following describes the proper procedural implementation based on these conditions?

A. Go to 1-OHP-4023-FR-I.2, and exit 1-OHP-4023-ES-1.2

B. Stay in 1-OHP-4023-ES-1.2, and implement 1-OHP-4023-FR-I.2, concurrently

C. Go to 1-OHP-4023-FR-Z.3, and exit 1-OHP-4023-ES-1.2

D.✓ Stay in 1-OHP-4023-ES-1.2, and implement 1-OHP-4023-FR-Z.3, concurrently

ANSWER: D

A. INCORRECT. Z.3 has a higher priority than I.2
B. INCORRECT. Z.3 has a higher priority than I.2
C. INCORRECT. Yellow path procedures are implemented concurrently.
D. CORRECT. Yellow path procedures are implemented concurrently. Z.3 has a higher priority than I.2

LESSON PLAN/OBJ: RO-C-EOP01/#22 & #23
REFERENCE: OHI-4023 Abnormal/Emergency Procedure User's Guide Attachment 5 sections 3.1 & 5.3, 2-OHP-4023-F-0.5, 2-OHP-4023-F-0.6

KA - 000061 2.4.4
Area Radiation Monitoring (ARM) System Alarms
Emergency Procedures/Plan
Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.
RO - 4.5 SRO - 4.7
CFR - 41.10 / 43.2 / 45.6
SCLR - 3SPK

KA Justification - Question requires candidate to determine that Z.3 is required entry based on Radiation levels and procedural implementation strategies.

Question Source - AUDIT RO23 088 (RO# N/A /SRO#086) (Bank-089-2)
Original KA - W/E16 EA2.1 (2.9/3.3)

86. 086 002/SRO/OK/MODIFIED/NRC EXAM 2004-030-1/006000 A2.12/4.5/4.8/H/3

A Loss of Off-Site Power has occurred on Unit 2. The crew is performing the actions of 2-OHP-4023-ES-0.2, Natural Circulation Cooldown.

Given the following plant conditions:

- RCS temperature is 537°F and trending down at approximately 25°F per hour.
- RCS pressure is 1183 psig and trending down slowly.
- East CCP is operating.
- 2-QRV-251, Charging Flow Control Valve, is fully open.
- 2-QRV-200, Charging Header Pressure Control Valve, is fully open.
- Pressurizer level is 15% and trending down slowly.
- High Steam Flow and Low Pressurizer Pressure SI signals are BLOCKED.

Which ONE of the following describes the correct action(s) for these conditions?

A.✔ Actuate Safety Injection and return to 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

B. Transition to 2-OHP-4023-ES-0.3, Natural Circulation Cooldown with a Steam Void in the Vessel.

C. Throttle open the steam dumps to raise the cooldown rate to 60°F/Hr IAW 02-OHP-4023-ES-0.2, Natural Circulation Cooldown.

D. Operate PZR heaters as necessary to stabilize RCS pressure and maintain subcooling greater than 25°F IAW 2-OHP-4023-ES-0.2, Natural Circulation Cooldown.

ANSWER: A

A. CORRECT. Subcooling is ~ 30°F (1183 psig + 15 psi = 1198 psia ~567°F sat-temp). 02-OHP-4023-ES-0.2 Foldout page directs this action when subcooling can NOT be maintained at >40°F

B. INCORRECT. Subcooling is already below the SI actuation setpoint and 02-OHP-4023-ES-0.3 is only made after step 13 if a rapid depressurization is required or if pressurizer level is high (indication of a void).

C. INCORRECT. The cooldown limit is 25°F/HR in 02-OHP-4023-ES-0.2

D. INCORRECT. The heaters would not function at <17% PZR level and the 25°F value is the cooldown limit NOT the subcooling requirement (

LESSON PLAN/OBJ: RO-C-EOP03/#23
REFERENCE: RO-C-EOP03, Plant Trips, Diagnosing Accidents, Natural Circulation Cooldown, E-0 Series EOPs, and Background Information pg. 89; 02-OHP-4023-ES-0.2 Natural Circulation Cooldown Foldout Page

KA - 006000 A2.12
Emergency Core Cooling System (ECCS)
Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Conditions requiring actuation of ECCS
RO - 4.5 SRO - 4.8
CFR - 41.5 / 43.5 / 45.3 / 45.13
SCLR - 3SPK

KA Justification - Question requires the candidate to identify a plant condition reflecting a loss of RCS level with automatic SI blocked, and the actions required.

Question Source - Cook NRC Exam 2004-030-1, RO24 Audit-086-21, 23137-SALEM02

Modified Stem to reduce Subcooling and raise PRZ level making Subcooling the requirement to actuate SI. Changed C & D distractors.

87. 087 065/SRO/OK/DIRECT/NRC2002-080 (#66/65)/010000 A2.02/3.9/3.9/H/2

Given the following plant conditions on Unit 2:

- Reactor power is 12%.
- The controlling Pressurizer (PRZ) Pressure Channel slowly fails high.
- The RO takes manual control of PRZ Pressure Master Controller and Lowers demand.
- All PRZ heaters have energized.
- RCS pressure is 2075 psig and slowly lowering.
- You notice that 2-NRV-163 (PRZ spray) is failed OPEN.
- When the RO places 2-NRV-163 in manual it will NOT close.

Which ONE of the following is the proper sequence of actions to stop the pressure reduction?

A. Trip RCP #23.
Dispatch an AEO to locally isolate Spray Valve 2-NRV-163.

B. Reduce Power to 8%.
Trip RCPs #23 and #24.
Dispatch an AEO to locally isolate Spray Valve 2-NRV-163.

C. Trip RCP #23.
Go to 2-OHP-4023-E-0, Reactor Trip Or Safety Injection.

D.✓ Manually trip the reactor.
Go to 2-OHP-4023-E-0, Reactor Trip Or Safety Injection.
Trip RCPs #23 and #24.

ANSWER: D

A. INCORRECT. Three loop operation is not allowed per the license, but plausible since the reactor would not trip at this power on the loss of a single RCP.

B. INCORRECT. Three loop operation is not allowed per the license. Plausible since 10% power also blocks 2 loop loss of flow trip.

C. INCORRECT. Per operating practices, the Reactor is tripped first and then the RCP. Plausible since a reactor trip and pump trip is required.

D. CORRECT. Three loop operation is not allowed. The RCP#3 & 4 must be stopped to stop the spray flow. Therefore the Reactor must be manually tripped and then the RCPs tripped.

LESSON PLAN/OBJ: RO-C-AOP-D6/RO-C-AOP0330412-E2
REFERENCE: 02-OHP-4024-207, Annunciator #207 Response: Reactor Coolant, Drop 61; 02-OHP-4023-ES-0-1, Reactor Trip Response

KA - 010000 A2.02
Pressurizer Pressure Control System (PZR PCS)
Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Spray valve failures
RO - 3.9 SRO - 3.9
CFR - 41.5 / 43.5 / 45.3 / 45.13
SCLR - 3SPK

KA Justification - Question requires candidate to identify actions required and their order based on PRZ Spray Valve Failure.

Question Source - RO24 AUDIT-079-11, Cook NRC Exam 2002-080 (#66/65), INPO - MODIFIED 19458

88. 088 289/SRO/OK/MODIFIED/NRC2007-29/007000 A2.02/2.6/3.2/H/3

Given the following plant conditions on Unit 1:

- Unit 1 was at 90% power.
- A pressurizer PORV was found to be leaking.
- The associated PORV block valve was shut 15 minutes later.
- All plant systems were in their normal lineup, with no alarms present in the Control Room.

For Information

1-OHP-4021-002-006, Pressurizer Relief Tank Operation

Attachment 3: Draining The Pressurizer Relief Tank
Attachment 4: Feed And Bleed Of PRT To Reduce Pressure Or Temperature
Attachment 7: Adjusting Pressure In The PRT
Attachment 10: Raising Level in the PRT

Which ONE of the following represents the expected status of the PRT and the actions that must be taken to restore it to normal limits?

A. PRT Temperature = 100°F, Level = 79%, and Pressure = 14 psig
Open the Vent to depressurize per Attachment 7 and then perform Attachment 10 to raise level.

B.✓ PRT Temperature = 130°F, Level = 81%, and Pressure = 8 psig
Open the Vent to depressurize per Attachment 7 and then perform Attachment 4 to lower temperature.

C. PRT Temperature = 260°F, Level = 82%, and Pressure = 21 psig
Perform Attachment 4 to lower pressure and temperature.

D. PRT Temperature = 240°F, Level = 95%, and Pressure = 3 psig
Reduce level per Attachment 3 and then perform Attachment 4 to lower temperature.

ANSWER: B

A. INCORRECT. The tank temperature and level are too low and the pressure is too high. The Vent will not open at this pressure.

B. CORRECT. PRT temperature is normally at Containment Temperature of ~100-110°F with level 80-84% and pressure of ~ 2-3 psig. The leaking PORV would have elevated PRT temperature & Pressure. The tank needs to be vented and a Feed & Bleed would be performed to cool the tank.

C. INCORRECT. Given this temperature and pressure the tank would be saturated. This is not expected to occur from a leaking PORV. The level would need to be much higher to reach this temperature.

D. INCORRECT. At this temperature pressure would need to be 10 psig. Level & Temperature would not be expected to increase this much.

LESSON PLAN/OBJ: RO-C-AOP-D2 / #RO-C-0020920412-E2
REFERENCE: 01-OHP-4022-002-009, Leaking Pressurizer Power Operated Relief Valve, 01-OHP-4021-002-006, Pressurizer Relief Tank Operation

KA - 007000 A2.02
Pressurizer Relief Tank/Quench Tank System (PRTS)
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:
Abnormal pressure in the PRT
RO - 2.6 SRO - 3.2
CFR - 41.5 / 43.5 / 45.3 / 45.13
SCLR - 2RI

KA Justification - Question presents PRT conditions following a leaking PORV and requires candidate to determine actions & procedural Attachments required to address high temperature & pressure.

Modified to Change to a leaking PORV from a lifted PORV. Modified Answer Values (Lowered Pressure to allow use of Vent in B) and changed actions in all distractors - Also included procedure attachments in stem & Choices.

Question Source - Cook NRC Exam 2007-29, NRC Exam 2002-100-1
Original KA - 007000 A1.02

89. 089 074/SRO/OK/DIRECT/MASTER 12AOPS0201-2/062000 2.4.47/4.2/4.2/H/3

Unit 1 was operating at a steady state power level of 48%.

The following alarms were lit due to the rate of rise over several days:

- Ann. 121, Drop 48, MAIN XFMR H2 CONC HI - HI
- Ann. 121, Drop 49, MAIN XFMR HYDRAN FAILURE OR H2 CONC HI.

At 13:35 it was noted that H_2 concentration was rising.
The H_2 concentration had previously been stable at 245 ppm.

The following Hydran readings have been taken:

Time	Reading
14:05	257 ppm
14:35	265 ppm
15:05	274 ppm
15:35	283 ppm
16:05	292 ppm
16:35	311 ppm

Based on CURRENT plant conditions, which ONE of the following actions should be directed by the Unit Supervisor?

Note: 1-OHP-4024-121, Drops 48 and 49, 1-OHP-4021-081-001, Data Sheet 1, are attached.

A. Continue monitoring, and notify Engineering of the abnormal readings.

B.✓ Reduce load by 10% in accordance with 1-OHP-4022-001-006, Rapid Power Reduction.

C. Reduce load by 30% in accordance with 1-OHP-4022-001-006, Rapid Power Reduction.

D. Trip the reactor and enter 1-OHP-4023-E-0, Reactor Trip or Safety Injection

ANSWER: B

A. INCORRECT. Action is required since rate of rise is greater than 20 ppm/hr. See answer B for details.
B. CORRECT. Rate of rise is less than 50 ppm/hr but greater than 20 ppm/hr. Action is to perform a 10% load reduction then evaluate in another 3 hours.
C. INCORRECT. Rate of rise is less than 50 ppm/hr but greater than 20 ppm/hr. Action is to perform a 10% load reduction then evaluate in another 3 hours.
D. INCORRECT. Shutdown is required if Hydran is greater than 750 ppm.

LESSON PLAN/OBJ: RO-S-AOP-D12/#RO-S-0810100801-T1
REFERENCE: 1-OHP-4024-121 Drops 48 & 49, 1-OHP-4021-081-001

Attachment Provided: 1-OHP-4024-121, Drops 48 and 49, 1-OHP-4021-081-001, Data Sheet 1

KA - 062000 2.4.47
A.C. Electrical Distribution System
Emergency Procedures/Plan
Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.
RO - 4.2 SRO - 4.2
CFR - 41.10 / 43.5 / 45.12
SCLR - 3SPR

KA Justification - Requires the ability to use the Annunicator Response procedure available in the control room to determine and direct the correct course of action based on a rising concentration of hydrogen in the main transformer.

Question Source - MASTER Bank 12AOPS0201-2

90. 090 005/SRO/OK/DIRECT/NRC EXAM 2004-025-3/008000 2.2.44/4.2/4.4/H/3

Unit 2 was operating at 100% power

The following alarms are received:

- Ann. 204:
  - Drop 88, WEST CCW SURGE TANK LVL HI OR LOW.
  - Drop 98, EAST CCW SURGE TANK LVL HI OR LOW.
- Ann. 207:
  - Drop 7, RCP #1 THERMAL BARRIER CLG WTR D/P HIGH.
  - Drop 8, RCP #1 THERMAL BARRIER CLG WTR TEMP HIGH.
  - Drop 9, RCP #1 THERMAL BARRIER DP LOW.
- Ann. 238:
  - Drop 10, R-17A EAST CCW HEADER HIGH RADIATION.
- Ann. 139:
  - 2-CRS-4301, East CCW Header High Radiation

Which ONE of the following statements is the required action and why?

The required actions are to verify CCW vent (2-CRV-412) shut, notify RP of high activity, and:

A. Enter 2-OHP-4022-002-001, Malfunction of a Reactor Coolant Pump, to address the #1 RCP seal failure.

B. Enter 2-OHP-4022-016-001, Malfunction of the CCW System, and monitor RCP Bearing temperatures since CCW lines in containment have ruptured.

C.✓ Enter 2-OHP-4022-016-003, CCW In-Leakage, to close the RCP thermal barrier valves (2-CCM-453 and 454) since the #21 RCP thermal barrier has failed.

D. Enter 2-OHP-4022-016-003, CCW In-Leakage, to remove letdown from service and place excess letdown in service since the letdown heat exchanger has failed.

ANSWER: C

A. INCORRECT. Incorrect - RCP Seal failure should not impact Surge tank level and temperature.

B. INCORRECT. CCW line rupture in Containment would NOT result in High CCW radiation.

C. CORRECT. Panel 207 Drops 7, 8, & 9 indicate a failure of the RCP Thermal Barrier. These alarms along with the others (Surge tank level and radiation) indicate the need to close the CCW from RCP Thermal Barrier Valves as per 02-OHP-4022-016-003 steps 1 & 2.

D. INCORRECT. Letdown would NOT cause Thermal barrier alarms.

LESSON PLAN/OBJ: RO-C-AOP-D3/RO-C-AOP0580412-E2
REFERENCE: 02-OHP 4022.016.003, CCW In-Leakage Procedure, RO-C-AOP-D3, Abnormal Operating Procedures - Day 3 TP-22 & 23

KA - 008000 2.2.44
Component Cooling Water System (CCWS)
Equipment Control
Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.
RO - 4.2 SRO - 4.4
CFR - 41.10 / 43.5 / 45.12
SCLR - 2RI

KA Justification - Question tests ability interpret indications of a CCW system problem and to determine the required procedure to be entered to address the problem.

Similar to Master Bank question 12AOPS0417-1, and NRC Exam 2004-025-3

91. 091 097/SRO/OK/MODIFIED/NRC2007-97/034000 A4.02/3.5/3.9/H/3

The plant is in MODE 6.
Fuel movement was suspended for repairs to the Spent Fuel Bridge Crane. Repairs to the Spent Fuel Bridge Crane are complete.

- Source Range Channels N31 and N32 are INOPERABLE.
- Source Range Channels N21 and N23 are OPERABLE.
- The West RHR pump has just been placed in service due to the failure of the East RHR pump seal.
- The Reactor Cavity Water Level is 644' 6".

The refueling team has established communications with the control room, and has requested permission to move the next fuel bundle from the fuel building to the core.

Are administrative conditions met to recommence fuel movement?

A. Yes, but only if the Reactor Cavity Water Level is raised to greater than 644' 9"

B. No, the East RHR pump must be restored to OPERABLE.

C✓ No, Source Range Channel N31 or N32 must be restored to OPERABLE.

D. Yes, provided that the operator selects N21 or N23 on the front of the Audio Count Rate Channel drawer.

ANSWER: C

A. INCORRECT. Level is only required to be 644' 1.5". 644' 9" is nominal.
B. INCORRECT. Only 1 RHR is required to be operable with > 23'.
C. CORRECT. For refueling to begin, 2 SR channels are required, one with an audible count rate indication. This comes only from N31 or N32. The gamma metrics (N21 or N23) may be used for the other channel. One RHR pump must be operating and level must be > 23' or 644' 1.5". The conditions for refueling would be met provided that the audible count rate is selected to N31.
D. INCORRECT. The gamma Metrics may be used for the second source range channel, but not selected for Audible counts.

LESSON PLAN/OBJ: RO-C-ADM13/ADM13.3.0
REFERENCE: 01-OHP-4030-127-037, Refueling Surveillance, Data Sheet 3

KA - 034000 A4.02
Fuel Handling Equipment System (FHES)
Ability to manually operate and/or monitor in the control room:
Neutron levels
RO - 3.5 SRO - 3.9
CFR - 41.7 / 45.5 to 45.8
SCLR - 2DR

KA Justification - Question requires candidate to determine if the minimum required Instrumentation is available to perform fuel movement. This includes the ability to monitor neutron levels through the audio count rate system.

Modified stem to make N32 Inoperable & added N21 as Operable. Changed distractor D (previous correct answer) to N21 or N23 and made Distractor C correct.

Question Source - Cook NRC Exam 2007-97
Original KA - Generic 2.2.26

92. 092 008/SRO/OK/NEW//072000 A2.01/2.7/2.9/H/3

Given the following plant conditions on Unit 1:

- The unit is operating at 100% power.
- A Containment Pressure Relief is in progress.
- VRS-1201 Upper Containment Normal Range Monitor failed due to a power supply failure.
- The Containment Pressure Relief is stopped.

Which ONE of the following describes the restrictions placed on Containment Purge/Pressure Relief Operations?

Note: Technical Specifications 3.3.6, Technical Requirements Manual 8.3.8, and the associated Bases are attached.

A. Containment Purge/Pressure Relief operations may NOT be performed until VRS-1201 is restored to operable status.

B✓ Containment Purge/Pressure Relief operations may continue under administrative controls provided that VRS-1201 is restored to operable status prior to entering Mode 4 following the next refueling outage.

C. Containment Purge/Pressure Relief operations may continue under administrative controls for up to 7 days, provided that area surveys of upper Containment are performed at least once every 24 Hrs.

D. Containment Purge/Pressure Relief operations may continue for up to 48 hours before VRS-1201 is required to be restored to operable status.

ANSWER: B

A. INCORRECT. Condition A of TS 3.3.6 allows Purge Operations if 2/3 channels per train are operable.
B. CORRECT. Condition A of TS 3.3.6 allows Purge Operations if 2/3 channels per train are operable. The failed channel must be fixed during the next refueling outage.
C. INCORRECT. Sampling every 24 hours is required per TRM 8.3.8 if BOTH VRS-1101 & 1201 are inoperable. Other Channels within TRM 8.3.8 have a 7 day limit.
D. INCORRECT. Purge operations are not limited. These requirements are from the Technical Specification 3.3.6 Condition C.

LESSON PLAN/OBJ: RO-C-01350\#10
REFERENCE: Technical Specifications 3.3.6, Technical Requirements Manual 8.3.8, and Bases

Attachment Provided: Technical Specifications 3.3.6, Technical Requirements Manual 8.3.8, and Bases

KA - 072000 A2.01
Area Radiation Monitoring (ARM) System
Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Erratic or failed power supply
RO - 2.7 SRO - 2.9
CFR - 41.5 / 43.5 / 45.3 / 45.13
SCLR - 3SPR

KA Justification - Question requires candidate to determine required Technical Specifications actions (use procedures) required due to the loss of an area radiation monitor power supply.

93. 093 090/SRO/OK/DIRECT - REPEAT/NRC2007-90/056000 2.4.45/4.1/4.3/H/3

Given the following plant conditions on Unit 2:

- The unit is at 50% power.
- East and West Main Feed Pumps (MFPs) are running.
- North and South Condensate Booster Pumps (CBPs) are running.
- Middle Condensate Booster Pump (CBP) is in Auto.

The following alarm is received in the Main Control Room:

- Ann. 216, Drop 82, CNDST BOOSTER PUMP MOTOR OVERHEATED.

While addressing the alarms, the following events occur:

- Ann. 216, Drop 72, CNDST BOOSTER PUMP MOTOR OVERLOAD TRIP - LIT.
- Ann. 216, Drop 73, CNDST BOOSTER PUMP DISCH PRESSURE LOW - LIT.
- Ann. 215, Drop 41, FEEDPUMP SUCTION HEADER PRESSURE LOW alarmed for approximately 3 seconds then cleared.

The following breaker indicating light conditions exist:

- North CBP: Red
- Middle CBP: Green
- South CBP: Green

Procedurally, the Unit Supervisor will direct the BOP to ______(1)________, and locally have an operator _______(2)___________.

A. 1) trip one Main Feedwater Pump

2) close the South CBP recirculation valve manual isolation.

B.✓ 1) start the Middle CBP

2) check the position of 2-CRV-224, Low Pressure Heater Bypass Valve

C. 1) start the Middle CBP

2) verify CBP recirculation valve manual isolation valves are throttled.

D. 1) trip one Main Feedwater Pump

2) verify open 2-CRV-224, Low Pressure Heater Bypass Valve

ANSWER: B

A. INCORRECT. Do not trip one running Main Feedwater Pump. Recirc valve is not isolated.
B. CORRECT. The Middle CBP should have started in Auto when Ann 216, Drop 73 alarmed. The Feedpumps do not trip until 180 psig for greater than 5 seconds. Ann. 215 Drop 41 is set at 188 psig. Momentary alarm on Ann. 215, Drop 41 may open CRV-224. If pressure remains above 188 psig. Valve should be closed.
C. INCORRECT. Check the position of CRV-224, NOT the recirc valve manual isolations.
D. INCORRECT. Do not trip one running Main Feedwater Pump. CRV-224 may open but should be re-closed under these conditions.

LESSON PLAN/OBJ: RO-C-05400/#8, #9, RO-C-05500/#11
REFERENCE: RO-C-05400, SOD-05500-001, 2-OHP-4024-215, Drops 41, 2-OHP-4024-216- Drop 72 & 73

KA - 056000 2.4.45
Condensate System
Emergency Procedures/Plan
Ability to prioritize and interpret the significance of each annunciator or alarm.
RO - 4.1 SRO - 4.3
CFR - 41.10 / 43.5 / 45.3 / 45.12
SCLR - 2RI

KA Justification - Question requires candidate to prioritize and determine significance of alarms and provide directions based on indications.

Question Source - COOK NRC Exam 2007-90, INPO # 22973 Prairie Island Unit 2 - 8/16/2002
Original Quest. KA - ..056000.A2.04

94. 094 003/SRO/OK/DIRECT- REPEAT/2006 NRC EXAM-088-3/194001 2.1.7/4.4/4.7/F/3

The following conditions exist :
-Large Break LOCA is in progress
-Containment pressure is 1.9 psig and stable
-You notice an ORANGE condition on CONTAINMENT CSF ST due to the "FLOOD LEVEL" lights being Lit.

What action will be directed by OHP-4023-FR Z.2, Response to Containment Flooding, and what is the concern if these actions are not successful?

A. Divert RHR flow from the Containment Sump to the RWST to lower Containment Level. High water levels could result in critical components needed for plant recovery being damaged and rendered inoperable.

B. Identify and isolate the source of excess water using control board indications and Containment Sump samples. Water levels could reach the bottom of the reactor vessel resulting in thermal shock and vessel failure.

C.✔ Identify and isolate the source of excess water using control board indications and Containment Sump samples. High water levels could result in critical components needed for plant recovery being damaged and rendered inoperable.

D. Stop both containment spray pumps. Water levels could reach the bottom of the reactor vessel resulting in thermal shock and vessel failure.

ANSWER: C

A. INCORRECT. Water is not pumped out of containment using the RHR pumps.
B. INCORRECT. Water reaching the Reactor vessel would not cause thermal shock or vessel failure.
C. CORRECT. Containment design basis flood level takes into account the entire water contents of the RCS, RWST, Ice condenser ice bed melt, and SI accumulators, plus the added mass of a LOCA and a steam line or feedline break inside containment. NESW and CCW may be major contributors to exceeding "flood" level and causing a loss of equipment required for long term cooling.
D. INCORRECT. The CTS pumps are not stopped due to high level. Water reaching the Reactor vessel would not cause thermal shock or vessel failure.

LESSON PLAN/OBJ: RO-C-EOP13/#7
REFERENCE: OHP-4023-FR-Z-2, Response to Containment Flooding Step 1 & Background

KA - 194001 2.1.7
Generic
Conduct of Operations
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
RO - 4.4 SRO - 4.7
CFR - 41.5 / 43.5 / 45.12 / 45.13
SCLR - 1B

K/A Justification - Question requires ability to evaluate the impact of plant conditions based on indications determine the operations implications.

Question Source Cook 2006 NRC Exam-088-3: INPO # 27561 Prairie Island - 4/23/2004
Original Quest. KA - 00WE15 EA2.2, E15 EK1.3

95. 095 003/SRO/OK/MODIFIED/NRC EXAM 2004-114-2/194001 2.1.25/3.9/4.2/H/3

At 16:30, Unit 2 experienced a LOCA. After the recirculation sump suction valves were opened in 2-OHP-4023-ES-1.3, Transfer to Cold Leg Recirculation, Sump Blockage has been identified. You have proceeded to 2-OHP-4023-ECA-1.3, Sump Blockage Control Room Procedure, and are currently at Step 19, "Check if SI Can Be Terminated."

At 17:20, the following plant conditions exist:

- RCS pressure is 300 psig and slowly lowering.
- NR RVLIS is 76% and slowly lowering.
- WR RVLIS is 23% and slowly lowering.
- CETC Average 345°F.
- RWST level is 13% and lowering.
- Containment pressure is 6.5 psig and stable.
- All Centrifugal Charging and Safety Injection pumps are running.
- Both Containment Spray Pumps are currently running with their suction aligned for recirculation.

Which ONE of the following actions is required?

Note: 2-OHP-4023-ECA-1.3, Sump Blockage Control Room Procedure, section is attached.

A. Do NOT stop or throttle ECCS Pumps, continue attempts to makeup to the RCS.

B. Terminate SI. Stop both SI pumps and 1 CCP. Isolate BIT injection and restore normal charging flowpath.

C. Attempt to establish minimum flow from ECCS pumps and throttle BIT injection as required to obtain about 305 gpm of injection flow.

D✓ Attempt to establish minimum flow from ECCS pumps and throttle BIT injection as required to obtain about 388 gpm of injection flow.

ANSWER: D

A. INCORRECT. This would be correct if RCPs were running with WR RVLIS <26%. Step 13a RNO to step 19.
B. INCORRECT. This would be correct if Subcooling was acceptable. Steps 14-18
C. INCORRECT. This flowrate is based on 110 minutes (50 minutes plus 1 hour) since the trip.
D. CORRECT. Based on these conditions (NR RVLIS<67%, RCPs tripped on Phase B, with subcooling < 86^{o}F) ECCS flow should be reduced to the minimum required per Figure1 as IAW Step 19.b RNO. The time after trip is 50 minutes which is equal to ~388 gpm.

LESSON PLAN/OBJ: RO-C-EOP09/#45
REFERENCE: 02-OHP-4023-ECA-1.3, Sump Blockage Control Rom Procedure, pg. 21-30 and Figure 1, pg. 49

Attachment Provided: 02-OHP-4023-ECA-1.3, Sump Blockage Control Room Procedure, pages 21-30 and Figure 1 (pg. 49)

KA - 194001 2.1.25
Generic
Conduct of Operations
Ability to interpret reference materials, such as graphs, curves, tables, etc.
RO - 3.9 SRO - 4.2
CFR - 41.10 / 43.5 / 45.12
SCLR - 3SPR

KA Justification -Question Requires candidate to use procedure & figure to determine required flow.

Question Source - Cook NRC Exam 2004-114-2, Master 12EOPC0932-4

Changed from procedure ECA-1.1 to ECA-1.3 (different procedure). Changed the times to make answer D correct rather than answer C.

96. 096 027/SRO/OK/DIRECT/POINTB-2006-30345/194001 2.2.35/3.6/4.5/H/2

At 0600, the following conditions are noted:

- Unit 1 is shutdown, preparing for refueling.
- Preparations are in progress for loosening the head bolts.
- Initial RCS temperature was 175°F.
- Initial RCS pressure was 100 PSIG.
- Normal Cooldown Alignments.
- Subsequently, RHR is lost and the RCS heats up at 4°F/minute.

Which ONE of the following correctly identifies the initial MODE and MODE at 0640?

	Initial MODE	**MODE at 0640**
A.✓	MODE 5	MODE 4
B.	MODE 6	MODE 5
C.	MODE 6	MODE 3
D.	MODE 5	MODE 3

ANSWER: A

A. CORRECT. With RCS Pressure at 100 PSIG, the RX Vessel head is still tensioned and with temperature is <200°F the plant is in mode 5. At 4°F/min for 40 minutes, temperature will rise 160°F final temperature will be 335°F which is Mode 4.

B. INCORRECT. Incorrect, MODE 6 is closely associated with refueling operations, since the stem states that preps for refueling are underway, the examinee may wrongly assume that MODE 6 has been entered. If this error is made and the examinee recognizes that a mode change has occurred, then MODE 5 would be a logical step up from MODE 6.

C. INCORRECT. Incorrect, combination of errors for B and D.

D. INCORRECT. Incorrect, if examinee recognizes that MODE 5 is the starting MODE, yet incorrectly assigns an incorrect value for the MODE change (LTOP temperature is 266°F) from 4 to 3, then this choice would be selected.

LESSON PLAN/OBJ: RO-C-TS01/#9
REFERENCE: Unit 1 Technical Specifications Table 1.1-1

KA - 194001 2.2.35
Generic
Equipment Control
Ability to determine Technical Specification Mode of Operation.
RO - 3.6 SRO - 4.5
CFR - 41.7 / 41.10 / 43.2 / 45.13
SCLR - 3PEO

KA Justification - Question requires candidate to determine current mode and projected mode based on temperature change.

Question Source -INPO Bank #30345 - POINT Beach -1202006 (2005 SRO Exam Q3)
Original Question KA - 000025.2.1.22

97. 097 067/SRO/OK/MODIFIED/NRC EXAM 2004 118-2/194001 2.3.5/2.9/2.9/H/3

A LOCA that resulted in significant core damage occurred at 1600 hours. Containment Pressure and Radiation levels were recorded as follows:

Time	**Radiation (R/Hr)**	**Pressure (psig)**
1600	400,000	6.2
1630	400,000	6.2
1700	400,000	5.6
1730	300,000	5.2
1800	300,000	4.5
1830	90,000	4.0

At 1835 hours, while performing Emergency Operating Procedures, a step is encountered which states "Check PRZ level - GREATER THAN 20% [24% ADVERSE]".

Which ONE of the following describes the required Pressurizer level and why?

A.✓ 20% because adverse values are no longer required because of the limited integrated dose and pressure reduction.

B. 24% because adverse values must be used until evaluated for lasting effects because the integrated dose limit has been exceeded.

C. 24% because adverse containment exists due to the current containment radiation dose rate.

D. 24% because adverse containment exists due to the current containment pressure.

ANSWER: A

A. CORRECT. Adverse containment values are required to be used when containment pressure is >5 psig or >10^5 R/Hr. When pressure lowers to <5 psig normal values may be used as long as the integrated dose is <10^6 R. The integrated dose is low enough to allow normal values to be used.

B. INCORRECT. The integrated dose is low enough to allow normal values to be used.

C. INCORRECT. The current Dose Rate is <10^5 R/Hr.

D. INCORRECT. Pressure is <5 psig.

LESSON PLAN/OBJ: RO-C-EOP01/#8 & #9
REFERENCE: OHI-4023, Abnormal / Emergency Procedure User's Guide, Attachment 2, Step 6, RO-C-EOP01

KA - 194001 2.3.5
Generic
Radiation Control
Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personal monitoring equipment, etc.
RO - 2.9 SRO - 2.9
CFR - 41.11 / 41.12 / 43.4 / 45.9
SCLR - 2RI

KA Justification - Ability to use rad monitor readings to determine the condition of containment for using adverse containment values while implementing the EOPs.

Question Source - Cook NRC Exam 2004 118-2(SRO87)

Modified question by lowering dose rates so that the integrated dose becomes $<10^5$ R changing the correct answer to A.

98. 098 001/SRO/OK/DIRECT/01ADMC1005-1/194001 2.3.6/2.0/3.8/H/3

While preparing a release permit for a waste monitor tank, it is determined that RFS-1010 (Liquid Waste Effluent Sample Flow) switch failed HIGH.
Repairs will take at least 3 days.

Which ONE of the following actions is required by PMP-6010-OSD-001, Off-site Dose Calculation Manual, regarding the liquid waste release?

Note: PMP-6010-OSD-001, Off-site Dose Calculation Manual, Attachment 3.2 is attached.

The release:

A. may NOT be approved until the flow monitor is restored to OPERABLE.

B. may be approved since the effluent monitor (RRS-1001) is OPERABLE.

C. may be approved for up to 30 days provided the flow rate is estimated at least once per 4 hours during the actual release.

D✓ may be approved after at least two independent samples are analyzed and at least two qualified persons independently verify the discharge valve lineup.

ANSWER: D

A. INCORRECT. A release may continue if 2 samples are taken and the flowpath is dual verified.
B. INCORRECT. RFS-1010 is required for Operability of RRS-1001.
C. INCORRECT. These are the required steps for Action 4.
D. CORRECT. A release may continue if 2 samples are taken and the flowpath is dual verified.

LESSON PLAN/OBJ: SR-O-SRO1/Task 0060220103
REFERENCE: PMP-6010.OSD.001 Att 3.2 ; 12-OHP- 4021.006.004 Att 3

Attachment Provided: PMP-6010-OSD-001, Off-site Dose Calculation Manual, Attachment 3.2

KA - 194001 2.3.6
Generic
Radiation Control
Ability to approve release permits.
RO - 2.0 SRO - 3.8
CFR - 41.13 / 43.4 / 45.10
SCLR - 3SPK (3SPR)

KA Justification - Question requires candidate to demonstrate ability to determine requirements to approve a release with an inoperable flow instrument.

Question Source - Master Bank - 01ADMC1005-1

99. 099 247/SRO/OK/DIRECT/MASTER 12EPPC0304-5/194001 2.4.38/2.4/4.4/F/3

Which ONE of the following sets of duties MAY be delegated by the Site Emergency Coordinator?

1) Notification of Plant Personnel.
2) Performing mitigating actions as required by the EOPs.
3) Classification of the Emergency.
4) Performing initial Offsite Dose Assessment.
5) Approving Protective Action Recommendations.
6) Directing the notification of Off-Site Officials.

A.✓ 1, 2, 4

B. 1, 3, 5

C. 2, 4, 6

D. 3, 5, 6

ANSWER: A

A. CORRECT. Notification of PLANT personnel, Direction of Mitigating actions, and Performance of Dose Assessment activities may be performed by others.
B. INCORRECT. Classification of the event and PAR approval may NOT be delegated.
C. INCORRECT. Directing Notification of Off-site officials may NOT be Delegated.
D. INCORRECT. These are the Duties that may NOT Be Delegated.

LESSON PLAN/OBJ: ST-C-EP03/#4
REFERENCE: PMP 2080.EPP.100, Emergency Response Sec 3.1.3

KA - 194001 2.4.38
Generic
Emergency Procedures/Plan
Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.
RO - 2.4 SRO - 4.4
CFR - 41.10 / 43.5 / 45.11
SCLR - 1P

KA Justification - Questions tests ability of candidate to identify which actions/activities must be performed by the emergency coordinator vs. those that can be delegated.

Question Source - Master Bank 12EPPC0304-5

100. 100 172/SRO/OK/NEW//194001 2.4.27/3.4/3.9/F/2

Which ONE of the following describes who has the primary responsibility to assume the role as the Consultant and Liaison to the Fire Brigade leader in the event of a fire emergency on Unit 2?

A. Shift Manager

B. Shift Technical Advisor

C. Unit 1 Senior Reactor Operator

D✓ Work Control Center Senior Reactor Operator

ANSWER: D

A. INCORRECT. The WCC SRO or designee acts as the liaison.
B. INCORRECT. The WCC SRO or designee acts as the liaison.
C. INCORRECT. The WCC SRO or designee acts as the liaison.
D. CORRECT. The WCC SRO or designee acts as the liaison.

LESSON PLAN/OBJ: RO-C-ADM01/#1, RO-C-ADM14/ADM14-18
REFERENCE: OHI-4000 Conduct of Operations: Standards, Attachment 23 Shift Staffing

KA - 194001 2.4.27
Generic
Emergency Procedures/Plan
Knowledge of "fire in the plant" procedure.
RO - 3.4 SRO - 3.9
CFR - 41.10 / 43.5 / 45.13
SCLR - 1P

KA Justification - Question requires knowledge of SRO responsibility as outlined in Shift Staffing during event of a fire.

SRO - DC Cook ILT NRC Exam 2008

#	ID	Answers 0
1	001 2	C
2	002 25	A
3	003 2	C
4	004 4	D
5	005 4	D
6	006 5	B
7	007 4	B
8	008 2	C
9	009 2	B
10	010 38	A
11	011 4	B
12	012 4	A
13	013 3	B
14	014 6	C
15	015 68	A
16	016 2	B
17	017 1	B
18	018 2	D
19	019 212	B
20	020 45	C
21	021 1	B
22	022 144	D
23	023 76	A
24	024 96	A
25	025 4	D
26	026 4	B
27	027 26	A
28	028 3	A
29	029 3	D
30	030 330	C
31	031 24	B
32	032 2	C
33	033 4	A
34	034 5	C
35	035 5	B
36	036 283	A
37	037 6	A
38	038 1	B
39	039 8	D
40	040 36	C
41	041 293	B
42	042 6	C
43	043 7	D
44	044 58	C
45	045 1	D
46	046 21	D
47	047 86	D
48	048 2	B
49	049 173	B
50	050 2	B

#	ID	Answers 0
51	051 1	A
52	052 70	C
53	053 1	C
54	054 39	A
55	055 1	B
56	056 4	B
57	057 2	A
58	058 58	C
59	059 1	D
60	060 1	B
61	061 13	B
62	062 62	A
63	063 7	A
64	064 11	B
65	065 3	A
66	066 21	B
67	067 71	A
68	068 76	B
69	069 36	D
70	070 65	B
71	071 1	A
72	072 4	D
73	073 1	B
74	074 59	C
75	075 12	B
76	076 121	A
77	077 8	D
78	078 7	A
79	079 95	C
80	080 15	B
81	081 14	D
82	082 2	D
83	083 27	C
84	084 7	B
85	085 2	D
86	086 2	A
87	087 65	D
88	088 289	B
89	089 74	B
90	090 5	C
91	091 97	C
92	092 8	B
93	093 90	B
94	094 3	C
95	095 3	D
96	096 27	A
97	097 67	A
98	098 1	D
99	099 247	A
100	100 172	D
SECTION 1 (100 items)		**100.00**