1. 001 001/BOTH/OK/NEW/NEW/000008 AK2.01/2.7/2.7/H/2

Given the following conditions on Unit 2:

- The crew is responding to a Reactor Trip and has transitioned to 2-OHP-4023-ES-0.1, Reactor Trip Response, from 2-OHP-4023-E-0, Reactor Trip or Safety Injection.
- Following the transition the crew notes the following conditions:
  - RCS Pressure is 2100 psig and lowering
  - 2-NRV-152, indicating lights show an intermediate position

Which ONE of the following describes the actions the operator should take to address these conditions:

A.✓ Close 2-NMO-152, PORV Block Valve, to stop discharge into the PRT.

B. Open Pressurizer Spray valves to depressurize the RCS and limit the loss of Reactor Coolant.

C. Turn on all Pressurizer Heaters to maintain RCS pressure.

D. Isolate air to containment to fail the Pressurizer PORV closed and stop the RCS mass loss.

ANSWER: A

A - CORRECT. Block valve is in series with the Pressurizer PORV. Closing the block valve will isolate the leak and prevent further depressurization of the RCS.

B - INCORRECT. Mass loss is occurring. Need to isolate the leak by isolating the PORV. Mass loss will continue as long as there is pressure in the RCS and the PORV is leaking.

C - INCORRECT. It is highly unlikely that the heaters can maintain pressure with a stuck open Pressurizer PORV. Additionally, even if the heaters could stabilize/raise pressure, the RCS leak would still persist.

D - INCORRECT. NRV-152 has Nitrogen Backup supply, so isolating air to containment will not force the PORV to fail closed.

LESSON PLAN/OBJ: RO-C-00202/#RO-C-00202-E9
REFERENCE: SOD-00202-001

KA - 000008 AK2.01
Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following:
Valves
RO - 2.7 SRO - 2.7
CFR - 41.7 / 45.7

KA Justification - Requires the knowledge of the interrelationship between the stuck open PZR PORV and the block valve during a vapor space leak. One block valve is in series with each PORV to allow isolation of a leaking, stuck open PORV.

Original Question # - NEW
Original Question KA - NEW

2. 002 012/BOTH/OK/DIRECT/RO24 AUDIT 76-12/000009 EA2.38/3.9/4.3/H/3

Given the following conditions on Unit 2:

- A reactor trip and safety injection have occurred.
- The crew is responding to a Small Break Loss of Coolant Accident (LOCA).
- All Reactor Coolant Pumps are tripped.
- The crew is depressurizing the Reactor Coolant System (RCS) in accordance with Step 13 of 2-OHP-4023-ES-1.2, Post LOCA Cooldown and Depressurization.
- A PORV is being used to depressurize the RCS.

As the depressurization occurs, which one of the following describes the expected trend of pressurizer level and the adverse operating condition that may initially occur as a result?

A. Lowering Pressurizer Level; Uncovering Pressurizer heaters.

B. Rising Pressurizer Level; Water solid conditions in the RCS and Pressurizer.

C.✓ Rising Pressurizer Level; Upper head region voiding may occur.

D. Lowering Pressurizer Level; Upper head region voiding may occur.

ANSWER: C

A - INCORRECT. Pressurizer will rise during depressurization versus lower. It is plausible that if the operator does not understand this concept and they believe Pressurizer level will drop that the heaters would become uncovered which is an undesirable condition.

B - INCORRECT. Although it is correct that Pressurizer level will rise and it is plausible that eventually the Pressurizer would go solid, that this would not INITIALLY occur.

C - CORRECT. The caution prior to commencing depressurization in ES-1.2 to refill the Pressurizer, states that a head void may occur as indicated by a rising Pressurizer level as water in transferred from the RCS to the Pressurizer.

D - INCORRECT. Pressurizer level will rise vs. lower. The reason is plausible and tests whether the student correctly understands the important concept.

LESSON PLAN/OBJ: RO-C-EOP09/#34 & 36
REFERENCE: 02-OHP-4023-ES-1.2 step 13
Background 12-OHP-4023-ES-1.2 -EOP Step #: 13 N1 ERG Step #: 11 C1

KA - 000009 EA2.38
Small Break LOCA
Ability to determine and interpret the following as they apply to a small break LOCA: Existence of head bubble
RO - 3.9 SRO - 4.3
CFR - 41.7 / 41.10 / 43.5 / 45.13
KA Justification - Requires ability to determine the indications that represent a bubble forming in the reactor vessel head during post LOCA cooldown and depressurization.

Original Question # - Cook RO24 Audit - 076-12
Original Question KA - EPE009.EA2.04

3. 003 002/BOTH/OK/MODIFIED/WATTSBAR - MAY2009/000011 EA1.05/4.3/3.9/F/3

Given the following conditions on Unit 1:

- A Large Break LOCA has occured.
- Safety Injection has actuated.

Which ONE of the following describes:

1) how the Centrifugal Charging Pump suction swaps to the RWST when a Safety Injection is initiated
-AND-
2) how the CHARGING PUMP SUCTION swapover to the containment sump is completed in accordance with 1-OHP-4023-ES-1.3, Transfer to Cold Leg Recirculation?

Note: VCT valves = QMO-451/QMO-452
RWST valves = IMO-910/IMO-911

A. 1) The VCT valves will start to close AFTER one of the RWST valves have traveled to the full open position.

2) The RWST valves will AUTOMATICALLY close after IMO-340, Charging Pp Suction from East RHR Hx has been opened.

B.✓ 1) The VCT valves will start to close AFTER one of the RWST valves have traveled to the full open position.

2) The RWST valves will be MANUALLY closed after IMO-340, Charging Pp Suction from East RHR Hx has been opened.

C. 1) The VCT valves will start to close AS SOON AS one of the RWST valves start to open.

2) The RWST valves will AUTOMATICALLY close after IMO-340, Charging Pp Suction from East RHR Hx has been opened.

D. 1) The VCT valves will start to close AS SOON AS one of the RWST valves start to open.

2) The RWST valves will be MANUALLY closed after IMO-340, Charging Pp Suction from East RHR Hx has been opened.

ANSWER: B

A - INCORRECT. Plausible, since the valves from the VCT starting to close when the valves from the RWST get fully open is correct.
B - CORRECT. The valves from the VCT will start to close when the valves from the RWST get fully open and after transfer to the containment sump the valves are placed in the A-Auto position.
C - INCORRECT. Plausible, because the ECCS valves used to swapover to the containment sump do travel together during the transfer as described in column (1). Handswitches positions is plausible since that is the normal position for the switches.
D - INCORRECT. Plausible, because the ECCS valves used to swapover to the containment sump do travel together during the transfer as described in column (1) and the handswitches being in A Auto position is correct.

LESSON PLAN/OBJ: RO-C-00300-E13, RO-C-EOP09/#22
REFERENCE: RO-C-00300 pg. 50, OHP-4023-ES-1.3, OP-2-98271

KA - 000011 EA1.05
Large Break LOCA
Ability to operate and/or monitor the following as they apply to a Large Break LOCA:
Manual and/or automatic transfer of suction of charging pumps to borated source
RO - 4.3 SRO - 3.9
CFR - 41.7 / 45.5 / 45.6

KA Justification - Stem conditions include a large break LOCA, and the question tests understanding of charging pump suction alignment during the event.

Original Question # - WattsBarMay2009
Original Question KA - 011 EA1.05

4. 004 006/BOTH/OK/DIRECT-REPEAT/NRC EXAM 2007-6/000015 AK1.04/2.9/3.1/H/3

Given the following conditions on Unit 2:

- Unit tripped from 29% power.
- RCP 21 breaker tripped open when the busses swapped.

Which one of the following describes the response of Thot and Tcold in Loop 1?

A. Tcold rises to approximately equal Thot.

B.✓ Thot lowers to approximately equal Tcold.

C. Tcold lowers, Thot remains approximately stable.

D. Thot rises, Tcold remains approximately stable.

ANSWER: B

A - INCORRECT. Tcold remains approximately the same, at low power near saturation for SG. Plausible due to lack of forced circulation prevents S/G from transferring heat to Main Steam Header assuming normal RCS flow direction, so RCS loop heats up to Thot

B - CORRECT. Loss of RCS flow in 1 loop, reverse flow in that loop will cause Thot to drop (no more forced circulation in that loop) to the Tcold value or slightly below.

C - INCORRECT. Tcold remains approximately the same, at low power near saturation for SG. Thot lowers since the core exit flow is not forced into the loop. Plausible due to reverse flow in RCS loop, from loss of RCP, allows Tcold to enter Steam Generator which removed additional energy thus lowering Tcold additionally.

D - INCORRECT. Thot lowers since the core exit flow is not forced into the loop. Plausible due to the Turbine Steam Demand did not change so overall Reactor Power will remain constant so the core will produce the same power from the remaining Steam Generators. Assuming the student believes normal RCS flowpath (at a reduced rate with no forced circulation) and if Tcold remains the same the only method to increase power from the remaining Steam Generators is to increase Thot.

LESSON PLAN/OBJ: RO-C-TRANS4\4A.4
REFERENCE: RO-C-TRANS4, RCS Loop Flow Transients

KA - 000015 AK1.04
017 Reactor Coolant Pump (RCP) Malfunctions
Knowledge of the operational implications of the following concepts as they apply to Reactor Coolant Pump Malfunctions:
Basic steady state thermodynamic relationship between RCS loops and S/Gs resulting from unbalanced RCS flow
RO - 2.9 SRO - 3.1
CFR - 41.8 / 41.10 / 45.3

KA Justification - Requires knowledge of the thermodynamic relationship of the temperature response of an idle loop following a malfunction of an RCP causing the RCP to trip.

Original Question # - NRC EXAM 2007-6, INPO # 23126 Salem Unit 1 - 11/4/2002
Original Question KA - 015.AA1.09

5. 005 024/BOTH/OK/DIRECT/RO26 AUDIT-24/000025 AA1.09/3.2/3.1/F/2

Given the following conditions on Unit 2:

- Reactor Coolant System (RCS) is in mid-loop condition
- The following indications are fluctuating on the running Residual Heat Removal (RHR): amps, flow, and discharge pressure

Which ONE of the following statements is correct regarding the standby RHR pump?

The standby RHR Pump should:

A.✓ NOT be immediately started because air entrainment could cause a loss of both RHR trains.

B. be immediately started because following a loss of RHR flow, an RCS pressurization may occur precluding gravity feed makeup.

C. be immediately started because under certain loss of RHR conditions, core uncovery or core voiding can occur within 15 to 20 minutes.

D. NOT be immediately started because starting an idle RHR pump under mid-loop conditions could cause an unacceptable reduction in reactor shutdown margin.

ANSWER: A

A - CORRECT. The ARG provides a clear guidance which includes industry experience of why operation of an RHR pump operating with air entrapment should be evaluated because it could lead to pump damage. Starting the other RHR pump could transfer the problem to the other pump leading to a complete loss of the system.

B - INCORRECT. Plausible because gravity makeup is a required action but starting the RHR pump is not an option immediately.

C - INCORRECT. While it is plausible that core uncovery or voiding could occur in a relatively short period of time, it is not correct that the RHR pump be started in this plant condition.

D - INCORRECT. Although it is correct that the pump should not be started it is not correct that SDM would be affected in this condition since it was already verified procedurally to meet TS plant conditions. Plausible since SDM is a concern during cooldown & RHR flow helps ensure proper mixing & SDM.

LESSON PLAN/OBJ: RO-C-AOP-D16/#RO-C-AOP0430412-E3
REFERENCE: OHP-4022-017-001

KA - 000025 AA1.09
Loss of Residual Heat Removal System (RHRS)
Ability to operate and/or monitor the following as they apply to the Loss of Residual Heat Removal System:
LPI pump switches, ammeter, discharge pressure gauge, flow meter, and indicators
RO - 3.2 SRO - 3.1
CFR - 41.7 / 45.5 / 45.6

KA Justification - Requires the ability to monitor RHR pumps amps and flow during mid-loop operations and determine based on the conditions what actions should be taken due to the abnormal conditions.

Original Question # - RO26 AUDIT-24
Original Question KA - APE035 AK2.02

6. 006 001/BOTH/OK/NEW/NEW/000027 2.2.40/3.4/4.7/H/3

Given the following conditions on Unit 1:

- Unit is operating at 100% power
- 1-NPP-151, PZR Press Channel 1, fails high
- The operator has taken manual control of Pressurizer pressure control and stabilized pressure at 2085 psig.

Following completion of the procedure for response to a malfunction of a pressurizer pressure instrument, what will be the status of the CVCS/Charging system?

The ______ CCP will be INOPERABLE with the associated emergency leakoff valve deenergized in the ______ position.

A.✓ East; open

B. East; closed

C. West; open

D. West; closed

ANSWER: A

A - CORRECT. Channel 1 affects QMO-225 for the East CCP. The Emergency leakoff is racked out in the open position to ensure minimum flow in the event of an SI.

B - INCORRECT. The Emergency leakoff is racked out in the OPEN position to ensure minimum flow in the event of an SI.

C - INCORRECT. Channel 1 affects QMO-225 for the East CCP, QMO-226 is for the West CCP from channel 2.

D - INCORRECT. Channel 1 affects QMO-225 for the East CCP, QMO-226 is for the West CCP from channel 2. The Emergency leakoff is racked out in the OPEN position to ensure minimum flow in the event of an SI.

LESSON PLAN/OBJ: RO-C-AOP-D6/#RO-C-AOP0330412-E3
REFERENCE: 1-OHP-4022-013-009, Pressurizer Pressure Instrument Malfunction

KA - 000027 2.2.40
Pressurizer Pressure Control (PZR PCS) Malfunction
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

KA Justification - Requires the operator to know the implications of a failed Pressurizer Pressure instrument on the operability of the CCPs and the required procedural actions to address TS concerns.

Original Question # - NEW
Original Question KA - NEW

7. 007 001/BOTH/OK/NEW/NEW/000029 EK2.06/2.9/3.1/H/3

Given the following conditions on Unit 2:

- The operators are implementing 2-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS
- Both Reactor Trip Breakers remain closed.
- The operators have completed steps of 2-OHP-4023-FR-S.1 through opening the MG Set output Breakers to shutdown the reactor.

Which ONE of the following describes the impact and potential consequences of the Reactor Trip Breakers remaining closed?

A. main steam line isolation signal will NOT occur to prevent excessive reactivity during the trip due to rapid RCS cooldown.

B.✓ feedwater isolation signal will NOT actuate to prevent excessive reactor coolant system cooldown from the overfeeding of the steam generators.

C. main generator trip signal will NOT be generated preventing transfer of busses to reserve feed.

D. feedwater flow conservation signal will NOT occur to ensure equal distribution of water to the steam generators.

ANSWER: B

A - INCORRECT. Main steam line isolation does not depend on the status of the Reactor Trip Breakers. This signal will occur as designed on appropriate containment pressure or SG parameters.

B - CORRECT. P-4 (Rx Trip Breaker Position) feeds the feedwater isolation signal. Either breaker being open will cause an isolation of flow to the SGs. However with neither breaker open, all FMOs and FRVs will remain open and Main Feedpumps will not trip. This will cause excessive flow to the SGs, which could lead to an overcooling of the RCS.

C - INCORRECT. Main Generator trip is caused by a sensed Turbine Trip (All Turbine Steam Valves Closed). There is no interlock between the Reactor Trip Breakers and the generator trip.

D - INCORRECT. Flow conservation is often confused with feedwater isolation. Flow conservation will start the Aux Feedwater Pumps and open the AFW Pump discharge valves. There is no interlock between the Reactor Trip Breakers and Flow Conservation.

LESSON PLAN/OBJ: RO-C-01100/#4
REFERENCE: RO-C-01100

KA - 000029 EK2.06
Anticipated Transient Without Scram (ATWS)
Knowledge of the interrelations between the ATWS and the following:
Breakers, relays, and disconnects
RO - 2.9 SRO - 3.1
CFR - 41.7 / 45.7

KA Justification - Requires knowledge of the interrelationship between an ATWS (Rx Trip Breakers NOT open) an the relays associated with feedwater isolation.

Original Question # - NEW
Original Question KA - NEW

8. 008 002/BOTH/OK/NEW/NEW/000051 AA1.04/2.5/2.5/H/3

Given the following conditions on Unit 1:

- The Unit is operating at 75% power with all systems in automatic.
- Main Turbine DCS is in "MW IN" in preparation for turbine valve testing.
- Condenser vacuum is lowering.

Assuming no action has been taken by the crew, which ONE of the following describes the response of the rod control system to this event?

Control rods will automatically:

A. insert due to the rise in Tavg from the rise in steam flow.

B. insert due to the rise in Tavg from the lowering in steam flow.

C.✓ withdraw due to the drop in Tavg from the rise in steam flow.

D. withdraw due to the a drop in Tavg from the lowering steam flow.

ANSWER: C

A - INCORRECT. Controls rods will withdraw. See Answer C. Plausible since during normal power escalations Tavg and Steam flow both rise.

B - INCORRECT. Controls rods will withdraw. See Answer C. Plausible due to design of system to insert rods on rising Tavg.

C - CORRECT. With the Main Turbine in "MW IN" control, the turbine valves are allowed to reposition to try to maintain Main Generator Load. As vacuum lowers, the turbine will become less efficient, causing more steam flow for the same MW output. As steam flow rises, RCS Tave will lower below Tref. With rods in AUTO, the rods will withdraw to minimize the Tave-Tref deviation.

D - INCORRECT. Steam flow will rise. See Answer C. Plausible since during normal power reductions Tavg and Steam flow both lower.

LESSON PLAN/OBJ: RO-C-05003/#RO-C-05003-E16, RO-C-01200/#RO-C-01200-E7
REFERENCE: SOD-01200-003, TS3000 Fig. 6-1

KA - 000051 AA1.04
Loss of Condenser Vacuum
Ability to operate and/or monitor the following as they apply to the Loss of Condenser Vacuum:
Rod position
RO - 2.5 SRO - 2.5
CFR - 41.7 / 45.5 / 45.6

KA Justification - Requires the ability to determine the proper operation of the control rod system during a transient caused by a Loss of Condenser Vacuum.

Original Question # - NEW
Original Question KA - NEW

9. 009 004/BOTH/OK/DIRECT/RO23 AUDIT-066-6/000054 AK3.04/4.4/4.6/F/3

Given the following conditions on Unit 1:

- Unit 1 was operating at 100% power when a condensate system transient caused both Main FW pumps to trip.
- The turbine and reactor failed to trip automatically.

In accordance with the immediate actions of 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS, the operators will:

1. Manually trip the Reactor, if it fails to trip insert control rods.
2. Manually actuate AMSAC.
3. Manually trip the Turbine, if it fails to trip, then runback the turbine.

Which ONE of the following describes the bases for these immediate actions in 1-OHP-4023-FR-S.1?

The safeguards systems are designed assuming that the only heat being added to the RCS is from _____________ . For an ATWS event with a loss of normal feedwater, a Turbine trip within 30 seconds will _____________________ .

A. fission product decay and RCP heat;
prevent challenging the Pressurizer PORV's.

B.✓ fission product decay and RCP heat;
maintain S/G inventory.

C. 5% power;
maintain S/G inventory.

D. 5% power;
prevent challenging the Pressurizer PORV's

ANSWER: B

A - INCORRECT. Turbine is tripped to maintain SG water inventory. Plausible due to first portion of answer is correct and second portion of answer is realistic possibility due to imbalance of energies from the Primary to the Secondary plants.

B - CORRECT. Per FR-S.1 Background Document, the assumed heat generation is from decay heat and RCP heat. Turbine is tripped to maintain SG water inventory.

C - INCORRECT. Per FR-S.1 Background Document, the assumed heat generation is from decay heat and RCP heat. Plausible due to common acceptance of 5% power being limitation of ESF system.

D - INCORRECT. Per FR-S.1 Background Document, the assumed heat generation is from decay heat and RCP heat. Turbine is tripped to maintain SG water inventory. Plausible due to common acceptance of 5% power being limitation of ESF system and second portion of answer is realistic possibility due to imbalance of energies from the Primary to the Secondary plants.

LESSON PLAN/OBJ: RO-C-EOP04/#15
REFERENCE: 01-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS, Step 1-3 Background, RO-C-EOP04 pg. 11

KA - 000054 AK3.04
Loss of Main Feedwater (MFW)
Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW):
Actions contained in EOPs for loss of MFW
RO - 4.4 SRO - 4.6
CFR - 41.5 / 41.10 / 45.6 / 45.13

KA Justification - Question tests knowledge of the reasons for FR S.1 Steps and actions associated with a Loss of FW ATWS.

Original Question # - RO23 AUDIT-066-6
Original Question KA - EPE 029 EK1.01

10. 010 002/BOTH/OK/DIRECT/PNTBEACH2002/000055 EK3.02/4.3/4.6/F/3

Given the following conditions on Unit 2:

- A loss of all AC Power occurred due to severe weather conditions and failure of emergency diesel generators to start and supply safeguard buses.
- The operating crew is carrying out actions of 2-OHP-4023-ECA-0.0, Loss of All AC Power.
- The operators are at a point where they are to commence cooldown and depressurization of the steam generators to 190 psig.

Based on these conditions, which ONE of the following statements describes the reason why a secondary depressurization is directed?

A. To prevent a challenge to the Core Cooling Safety Function Status Tree which is being monitored for implementation.

B. To remove stored energy in the steam generators to limit the potential of challenging RCS integrity.

C. To remove available energy in the steam generators and thus minimizing any challenges to the containment structure if a Faulted S/G were to occur.

D.✓ To minimize RCS inventory loss through the RCP seals, which maximizes time to core uncovery.

ANSWER: D

A - INCORRECT. Status Trees are monitored for information only. ECA 0.0 has built into the mitigating strategy to manage all actions addressed by FRP's as well as the FRP's are for information use only in ECA-0.0.

B - INCORRECT. SG PORVS should limit SG pressures but the primary concern is unrecoverable loss of RCS inventory. Plausible due to reduction in risk of having an RCS integrity challenge which is not a concern in this event

C - INCORRECT. SG PORVS should limit SG pressures but the primary concern is unrecoverable loss of RCS inventory. Plausible due to reduction in risk of having a challenge to the Containment structure due to a reduction in the available energy inside Containment

D - CORRECT. The primary concern is a loss of RCS inventory with no way to recover level. This could lead to core uncovery. The SGs are depressurized to lower RCS temperature and pressure to slow the loss of inventory.

LESSON PLAN/OBJ: RO-C-EOP14/#5 & 9
REFERENCE: 2-OHP-4023-ECA-0.0, Step 19 and Note 1 Background

KA - 000055 EK3.02
Loss of Offsite and Onsite Power (Station Blackout)
Knowledge of the reasons for the following responses as they apply to the Station Blackout:
Actions contained in EOP for loss of offsite and onsite power
RO - 4.3 SRO - 4.6
CFR - 41.5 / 41.10 / 45.6 / 45.13

KA Justification - Question tests knowledge of the reason the SGs are depressurized in ECA-0.0.

Original Question # - INPO # 20572 Point Beach 1 - 2/2/2002
Original Question KA - 055.EK3.02

11. 011 002/BOTH/OK/DIRECT-REPEAT/NRC EXAM 2007-041/000056 AK1.03/3.1/3.4/H/3

Given the following conditions on Unit 2:

- Unit has just tripped due to a Loss of Offsite power.
- Both EDGs started and energized the required loads.
- All equipment responded as designed.

<u>The following conditions exist:</u>
- Containment parameters are normal
- Average core exit thermocouple (CET) temperature is stable.

Which ONE of the following combination of RCS pressure and average CET temperature verifies the MINIMUM required subcooling to AVOID Safety Injection per 2-OHP-4023-ES-0.2, Natural Circulation Cooldown?

A. 600 psig, 590°F

B. 500 psig, 460°F

C. 450 psig, 430°F

D.✓ 375 psig, 400°F

ANSWER: D

A - INCORRECT. RCS is saturated - Tsat is 489°F
B - INCORRECT. RCS is 10°F subcooled - Tsat is 470°F
C - INCORRECT. RCS is 30°F subcooled - Tsat is 460°F
D - CORRECT. 2-OHP-4023-ES-0.2 , Foldout Page (FOP) needs >40°F of subcooling, or requires that a SI be actuated. Tsat fo 400 psia (375 psig + 15 psi) is 444.6°F. Based on the conditions provided, 44.6°F of subcooling exists, exceeding the 40°F requirement.

LESSON PLAN/OBJ: RO-C-EOP03/#18
REFERENCE: 2-OHP-4023-ES-0.2, Natural Circulation Cooldown Foldout Page, RO-C-EOP03, Plant Trips, Diagnosing Accidents, Natural Circulation Cooldown, E-0 Series EOPs, and Background Information pg. 89

Reference Provided: Steam Tables

KA - 000056 AK1.03
Loss of Offsite Power
Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power:
Definition of subcooling: use of steam tables to determine it
RO - 3.1 SRO - 3.4
CFR - 41.8 / 41.10 / 45.3

KA Justification - Requires the use of the steam table to determine subcooling during a loss of offsite power event on natural circ cooldown.

Original Question # - 21521-KEWAUNEE02, NRC2004-41-2

12. 012 003/BOTH/OK/MODIFIED/RO25 AUDIT-12/APE057 AA1.02/3.8/3.7/H/4

Given the following conditions on Unit 1:

- Unit is operating at 60% power
- Pressurizer Level Control is in MANUAL
- Pressurizer LEVEL CTRL SELECTOR switch is in the Channel 2-3 position
- CRID 3 power supply fails

Assuming no operator action, which ONE of the following statements describes the effect of this failure on the CVCS and PZR level control system?

A.✓ QRV-251, CCP Disch Flow Control fails OPEN
Letdown Isolates
Actual Pressurizer Level Rises

B. QRV-251, CCP Disch Flow Control fails CLOSED
Letdown Isolates
Actual Pressurizer Level Rises

C. QRV-251, CCP Disch Flow Control fails CLOSED
QRV-200, Charging Header Pressure Control Valve fails OPEN
Actual Pressurizer Level Lowers

D. QRV-251, CCP Disch Flow Control fails OPEN
QRV-200, Charging Header Pressure Control Valve fails CLOSED
Actual Pressurizer Level Lowers

ANSWER: A

A - CORRECT. Loss CRID 3 causes a Loss of PZR level Channel NLP-153 which will result in an indicated low Pressurizer level. This will cause the PZR Level control to Close QRV-112 and Open QRV-251. (Note that QRV-251 will also fail Open & QRV-112 will Close from Loss of CRID 3 per OHP-4021-082-008 Table 3. i.e. - either NLP-153 failing or loss of power causes same effects)

B - INCORRECT. Channel 3 will fail low. QRV-251 will fail open.

C - INCORRECT. Channel 3 will fail low. QRV-251 will fail open. Level will rise.

D - INCORRECT. Channel 3 will fail low. Level will rise.

LESSON PLAN/OBJ: RO-C-00300/#RO-C-00300-E9, RO-C-00202/#RO-C-00202-E14
REFERENCE: SOD-00202-003, Pressurizer Level Control;
1-OHP-4021-082-008

Modified: Changed to CRID 3 from 4 (Answer D to A) & Changed Distractor C.

KA - 000057 AA1.02
Loss of Vital AC Electrical Instrument Bus
Ability to operate and/or monitor the following as they apply to the Loss of Vital AC Instrument Bus:
Manual control of PZR level
RO - 3.8 SRO - 3.7
CFR - 41.7 / 45.5 / 45.6

KA Justification - Requires the ability to monitor the response of pressurizer level control to a failure of a CRID (AC Power) while operating in manual control. Student must be able to identify correct failure position for multiple valves failing in different directions and evaluate the impact on the Pressurizer level.

Original Question # - Audit RO22-BOTH-59 (#52), RO25 AUDIT-12
Original Question KA - KA SYS 004K2.06, APE057 AA1.02

13. 013 001/BOTH/OK/NEW/NEW/000058 2.4.20/3.8/4.3/F/2

Which ONE of the following describes the reason for disabling AMSAC, de-energizing DCS Inverters, and stopping all DC powered Turbine Oil Pumps in 2-OHP-4023-ECA-0.0, Loss of All AC Power ?

A. Allow turbine oil systems to be de-energized and drain to the main turbine lube oil tank.

B. Prevent inadvertent actuation of control systems and auto start of pumps.

C. ✓ Extend the DC battery life for N Train and BOP batteries.

D. Limit overheating of cabinets and pump motor overload.

ANSWER: C

A - INCORRECT. Plausible since oil systems will drain back to the main turbine LO tank and student may think this will help reduce risk of fire.
B - INCORRECT. Plausible due to concerns for smart shorts of equipment during emergencies.
C - CORRECT. ECA-0.0, Step 17 Note states that DC Loads are shed to extend the life of the DC batteries associated with the loads..
D - INCORRECT. Plausible since other control room cabinets are opened in ECA-0.0 to preclude overheating of instrumentation.

LESSON PLAN/OBJ: RO-C-EOP14/#11
REFERENCE: 2-OHP-4023-ECA-0.0, Step 17

KA - 000058 2.4.20
Loss of DC Power
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

KA Justification - Question tests operational implication (plant impact) of note in loss of DC procedure.

Original Question # - NEW
Original Question KA - NEW

14. 014 004/BOTH/OK/DIRECT/CALLAWAY2007-59/000059 AK1.02/2.6/3.2/F/3

Given the following conditions:

- An accidental spill of the Monitor Tank has occurred in the Aux Building.
- Radiation levels in the area of the spill are 40 mRem per hour at 30 cm.
- Contamination levels based on smear on the floor around the tank are 1.2 $x10^4$ dpm/100 cm^2 beta-gamma.

Which ONE of the following describes how the area will be posted in accordance with PMI-6010, Radiation Protection Plan?

A. Radiation Area **ONLY**.

B. Contamination Area **ONLY**.

C.✓ Radiation Area **AND** Contamination Area.

D. High Radiation Area **AND** Contamination Area.

ANSWER: C

A - INCORRECT. Greater than >1000 dpm/100 cm^2 is a contamination area. Plausible because this answer is only partially correct.
B - INCORRECT. Greater than 100 mrem/ hr is a high radiation area. Plausible because this answer is only partially correct.
C - CORRECT. This area should be posted as a radiation area (>5 mrem in 1 hour and <100 mrem/hr) and a contamination area (>1000 dpm/100 cm^2).
D - INCORRECT. Greater than 100 mrem/ hr is a high radiation area. Area is <100 mrem/ hr.

LESSON PLAN/OBJ: RO-C-RP02/#4 & 7
REFERENCE: PMI-6010, Section 4.7, RO-C-RP02

KA - 000059 AK1.02
Accidental Liquid Radwaste Release
Knowledge of the operational implications of the following concepts as they apply to Accidental Liquid Radwaste Release:
Biological effects on humans of various types of radiation, exposure levels that are acceptable for nuclear power plant personnel, and the units used for radiation-intensity measurements and for radiation exposure levels
RO - 2.6 SRO - 3.2
CFR - 41.8 / 41.10 / 45.3

KA Justification - Requires knowledge of the biological implications to workers related to the and postings requirements for an accidental release (spill of Monitor Tank contents) in the Aux Building. In addition, requires knowledge of the units and acceptable levels of radiation/contamination for these conditions.

Original Question # - CALLAWAY2007-59
Original Question KA - 000059 AK1.02

15. 015 005/BOTH/OK/NEW/NEW/000061 2.2.36/3.1/4.2/H/3

Given the following condition:

- 1-MRV-213, Unit 1 SG11 PORV, was locally isolated due to excessive leakby.

Which ONE of the following describes the status of the SG11 PORV Radiation Monitoring and the reason?

A. Channel MRA-1601 is Unavailable. Technical Requirement actions are NOT required since all of the other SG PORV monitors are operable.

B.✓ Channel MRA-1601 is Inoperable. Technical Requirements actions are required since PORV isolation renders the radiation monitoring function Inoperable.

C. Channel MRA-1601 is Unavailable. SG11 Radiation Monitoring is still Operable since MRA-1602 is still functioning.

D. Channel MRA-1601 is Operable. Isolation of the SG PORV does NOT affect radiation monitor operability.

ANSWER: B

A - INCORRECT. The SG PORV Radiation Monitor is declared inoperable if the PORV Is isolated. There is only one Rad Moniotr per steamline. Plausible since MRA-1601 and MRA-1602 are on the same monitor and/or if the canidate applies the 3/4 logic used in many instrumentation specifications.

B - CORRECT. TRO 8.3.8 requires 1 Channel per loop to be Operable. The SG PORV Monitor is required to be declared Inoperable if the PORV is closed or isolated. Declaring the SG PORV Radiation monitor inoperable requires that Function 2.b Action C be applied.

C - INCORRECT. The SG Radiation Monitor is declared inoperable if the PORV Is isolated. Plausible since MRA-1601 and MRA-1602 are on the same monitor.

D - INCORRECT. The SG PORV Radiation Monitor is declared inoperable if the PORV Is isolated. Plausible since MRA-1601 will still indicate but without auto PORV operation, the SG safeties may lift allowing unmonitored release path.

LESSON PLAN/OBJ: RO-C-05103/RO-C-05103-E4, RO-C-05103-E11
REFERENCE: TRM 8.3.8, RO-C-05103 pg. 13, 26-27, RO-C-AOP-D12 Slide 15, 1-OHP-4022-013-012 Step 4

KA - 000061 2.2.36
Area Radiation Monitoring (ARM) System Alarms
Equipment Control
Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.
RO - 3.1 SRO - 4.2
CFR - 41.10 / 43.2 / 45.13

KA Justification - Question tests ability to determine the status of LCO on PORV Rad Monitors due to maintenance activity on the PORV.

Original Question # - NEW
Original Question KA - NEW

16. 016 008/BOTH/OK/MODIFIED/NRC EXAM 2004-130-3/000065 2.4.8/3.8/4.5/F/3

Given the following conditions on Unit 2:

- Unit was operating at 100% power when a malfunction of the Control Air system occurs.
- The Control Air header rapidly depressurizes and cannot be restored.

Which ONE of the following describes the correct operator response?

Immediately trip the Reactor and implement:

A.✓ 2-OHP-4023-E-0, Reactor Trip or Safety Injection.
2-OHP-4022-064-002, Loss Of Control Air Recovery, may be performed concurrently after transitioning to 2-OHP-4023-ES-0.1, Reactor Trip Response.

B. 2-OHP-4023-E-0, Reactor Trip or Safety Injection.
2-OHP-4022-064-002, Loss Of Control Air Recovery, is NOT needed since the EOP network may be performed without reliance on Control Air.

C. 2-OHP-4023-E-0, Reactor Trip or Safety Injection.
2-OHP-4022-064-002, Loss Of Control Air Recovery, may NOT be performed until completion of 2-OHP-4023-ES-0.1, Reactor Trip Response.

D. 2-OHP-4022-064-002, Loss Of Control Air Recovery, until restoration of Control Air from any source.
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 Non-Accident (ES-0.1, 0.2 or 0.3) Emergency Procedures after the immediate actions are complete at US discretion.

B - INCORRECT. Performance of 02-OHP-4023-E-0 is required upon the reactor trip, but the operators must continue to perform 02-OHP-4022-064-002 to address the loss of Control Air.

C - INCORRECT. User's Guide allows Abnormal Procedures to be implemented concurrently with Non-Accident (ES-0.1, 0.2 or 0.3) Emergency Procedures.

D - INCORRECT. The Unit Supervisor should direct action of 02-OHP-4023-E-0, first, NOT as time allows. 02-OHP-4023-E-0 actions take priority over 02-OHP-4022-064-002.

LESSON PLAN/OBJ: RO-C-EOP01/#25
REFERENCE: OHI-4023 Abnormal/Emergency Procedure User's Guide, Attachment 2, Step 3.0.

KA - 000065 2.4.8
Loss of Instrument Air
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

K/A Justification - Requires the knowledge of how to use Loss of Control Air (Abnormal Operating Procedure) in conjunction with the Emergency Operating Procedures.

Question Source: NRC EXAM 2004-130-3

17. 017 001/BOTH/OK/DIRECT/CM-AS17-41643/000067 AK1.02/3.1/3.9/H/3

Given the following conditions:

- A Main Transformer fire has occurred on Unit 1.
- The Unit 1 Reactor and Main Turbine/Generator have been tripped.
- The Turbine AEO has reported that the Main Transformer deluge system has actuated.
- The Outside Tour AEO has reported that all three fire water pumps are running.

The reported status of the Fire Water System is ____________ for this event. The Main Transformer deluge system ___________ expected to automatically actuate once the Main Generator is tripped, ___________ fire water pumps are expected to be running.

A✓ abnormal; is; but only 2

B. abnormal; is not; but 3

C. normal; is; and 3

D. abnormal; is not; and only 2

ANSWER: A

A - CORRECT. For any given actuation of the fire system, the maximum number of pumps running should be 2. All three pumps running are an indication of a piping rupture. The deluge valve will not automatically actuate until the main transformer is de-energized.

B - INCORRECT. Deluge will automatically actuate. Only 2 pumps should be running. Plausible due to a fire requires activation of the deluge to put out as well as student must know all 3 pumps are not expected to be running.

C - INCORRECT. Condition is not normal. Only 2 pumps should start once transformer is de-energized. Plausible due to student must know all 3 pumps are not expected to be running.

D - INCORRECT. Deluge will automatically actuate. Plausible due to a fire requires activation of the deluge to put out as well as student must know only 2 pumps are expected to be running.

LESSON PLAN/OBJ: RO-C-AS17/#3
REFERENCE: 1-OHP-4024-101, ANNUNCIATOR #101 RESPONSE: PLANT FIRE SYSTEM, Drop 2

KA - 000067 AK1.02
Plant Fire on Site
Knowledge of the operational implications of the following concepts as they apply to Plant Fire on Site:
Fire fighting
RO - 3.1 SRO - 3.9
CFR - 41.8 / 41.10 / 45.3

KA Justification - Requires an operational knowledge of the normal configuration of fire fighting equipment during a normal actuation of the fire protection system.

Original Question # - CM-AS17 - 41643
Original Question KA - 086 A3.02

18. 018 004/BOTH/OK/NEW/NEW/000068 AK2.07/3.3/3.4/F/3

Given the following conditions on Unit 2:

- Reactor Power is at 100% when a fire occurs in the Control Room Cable Vault.
- A large amount of smoke accumulates in the Control Room,
- The Shift Manager determines that the main control room must be evacuated in accordance with 2-OHP-4025-001-001, Emergency Remote Shutdown.

Which ONE of the following describe the operation of the Emergency Diesel Generators (EDGs)?

A. Start both EDGs prior to control room evacuation.

Locally control EDGs from LSI panels as required.

B. Trip both EDG HEAs prior to leaving control room.

Restore EDGs per 2-OHP-4025-R-15, if required

C.✓ Leave EDG control switches as is.

Locally Trip and Isolate EDGs in accordance with 2-OHP-4025-LTI-3, if required.

D. Leave EDG control switches as is.

Locally control EDGs from LSI panels as required.

ANSWER: C

A - INCORRECT. EDGs are isolated after control room evacuation is complete. There is no local control for EDGs from the LSI panels. Plausible to prepare the EDG's for loading when needed and provides a logical place for control of the EDG outside of the Control Room

B - INCORRECT. EDGs are left as is. Trip, Isolation, and restoration is performed following evacuation per appendix R procedures. Plausible to prevent inadvertent operation of the EDG's complicating management of power sources

C - CORRECT. EDGs are left as is. Trip, Isolation, and restoration is performed following evacuation per appendix R procedures.

D - INCORRECT. See B. Additionally, there is no local control for EDGs from the LSI panels. Plausible because it provides a logical place for control of the EDG outside of the Control Room

LESSON PLAN/OBJ: RO-C-EC02\#4
REFERENCE: 2-OHP-4025-001-001 Steps 1-15, 2-OHP-2025-LTI-3

KA - 000068 AK2.07
Control Room Evacuation
Knowledge of the interrelations between the Control Room Evacuation and the following:
ED/G
RO - 3.3 SRO - 3.4
CFR - 41.7 / 45.7

KA Justification - Question addresses how the EDG operation is addressed during a Control Room Evacuation due to fire.

Original Question # - NEW
Original Question KA - NEW

19. 019 004/BOTH/OK/DIRECT/NRC EXAM 2004-051-3/000077 AA2.07/3.6/4.0/H/3

Given the following conditions on Unit 1:

- Unit 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 buses drops.

The following conditions now exist:

- T11A Voltage Indication is 112 Volts
- T11B Voltage Indication is 114 Volts
- T11C Voltage Indication is 113 Volts
- T11D Voltage Indication is 114 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. T11A and T11C busses will be energized by their respective EDG.

C.✓ T11A and T11B busses will be energized by its respective EDG.

D. Only T11A bus will be energized by its respective EDG.

ANSWER: C

A - INCORRECT. T11 C and T11D will NOT deenergize since T11D is > 113V. Plausible if setpoint not known by student

B - INCORRECT. T11C will NOT deenergize since T11D is > 113V. Plausible due to actual setpoint known by student but alignment not understood by student for bus stripping.

C - CORRECT. An Undervoltage condition of 113 V will energize 62-1 T11A. After a 111 Second delay it will open T11A9 and T11B1 causing T11 A and T11B to lose power. This will cause the EDG to start and energize T11A and T11B.

D - INCORRECT. T11B will also recieve a trip signal and be energized by the EDG. Plausible if alignment not understood by student for bus stripping.

LESSON PLAN/OBJ: RO-C-08201/#6
REFERENCE: RO-C-08201, Engineered Safety Systems Electrical pg. 29, 32-33, and Att.3. Annunciator #121 Response, Drop 78 Train B Aux Buses Undervoltage pg. 178-182, SOD-08201-001

KA - 000077 AA2.07
Generator Voltage and Electric Grid Disturbances
Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances:
Operational status of engineered safety features
RO - 3.6 SRO - 4.0
CFR - 41.5 and 43.5 / 45.5 / 45.7, and 45.8

KA Justification - Requires determination of the status of electrical buses following an electrical grid disturbance resulting in degraded grid voltage.

Original Question # - NRC EXAM 2004-051-3

20. 020 002/BOTH/OK/DIRECT/RO24 AUDIT-12/003000 K2.02/2.5/2.6/H/3

Given the following conditions on Unit 1:

- Reactor power is 100%.
- West CCP is in operation with the East CCP in standby.
- West CCW pump is tagged out for maintenance.

Which ONE of the following describes the immediate operator actions required for a loss of Bus T11D:

A✔ Trip reactor because the RCP seals will overheat without Component Cooling flow.

B. Trip reactor because there is NO charging flow to replace letdown.

C. Initiate a controlled shutdown because the Charging pump will overheat without Component Cooling flow.

D. Initiate a controlled shutdown because the RCP seals will overheat without charging flow.

ANSWER: A

A - CORRECT. Per Loss of CCW procedure, Trip Reactor and Then trip RCPs.
B - INCORRECT. Plausible since Letdown is isolated to conserve level, a trip is not required due to loss of letdown.
C - INCORRECT. Plausible since CCPs must be shutdown but shutdown is required within 1-2 minutes so a controlled Shutdown is not warranted.
D - INCORRECT. Plausible since RCP seals will overheat when charging is stopped. An attempt is made to crosstie to the opposite unit. The concern with RCP motor bearings is more severe and requires immediate trip.

LESSON PLAN/OBJ: RO-C-01600 / #RO-C-01600-E6,
RO-C-AOP-D14\#RO-C-AOP0560412-E3
REFERENCE: 1-OHP-4022-016-004, Loss of Component Cooling Water,
RO-C-AOP-D14, RO-C-01600 pg. 13-14

KA - 003000 K2.02
Reactor Coolant Pump System (RCPS)
Knowledge of bus power supplies to the following:
CCW pumps
RO - 2.5 SRO - 2.6
CFR - 41.7

KA Justification - Requires the knowledge of the bus power supplies to the CCW pumps and the effect a loss of the CCW pumps will have on the RCPs.

Original Question # - INPO - DIRECT 20154, COOK02-052-1,R39,S45, RO24 Audit-12
Original Question KA - 003000-K2.02, 062.A2.01

21. 021 004/BOTH/OK/DIRECT/RO26 AUDIT-4/004000 2.4.47/4.2/4.2/H/3

Given the following conditions on Unit 2:

- Reactor is at 100% power.
- All control systems are in normal alignment.
- Letdown flow is aligned with a flow of 120 gpm at QFI-301.

The following parameters are now noted on the CVCS system:

- Seal Return Flows are 3 gpm per RCP
- Charging flow is 137 gpm and rising.
- 2-QTA-160, Regen HX Outlet Temp - Letdown, has lowered 5°F from its steady state value.
- VCT level is 33% and lowering.
- PZR level is 55% and lowering slowly.
- RCS temperature is 574°F and stable.

Which ONE of the following describes the effect on the unit and the action required to address the conditions?

A. RCS leakage is from the letdown line between the orifices and the letdown containment isolation valves. Isolate Letdown.

B.✓ RCS leakage is from the charging line on the RCS side of the regenerative heat exchanger. Isolate Charging and Letdown.

C. RCS leakage is from the letdown line on the CVCS side of the regenerative heat exchanger. Initiate an investigation to determine if the leak is isolable.

D. RCS leakage is from the charging line on the CVCS side of the regenerative heat exchanger. Isolate Charging and Letdown.

ANSWER: B

A - INCORRECT. Leak is in the charging header. This leakage would cause a rise in letdown flow through the Regen HX and a rise on 2-QTA-160, Regen HX Outlet Temp - Letdown.

B - CORRECT. If Regen Hx Outlet temperature is lowering, then more charging flow is going through the Regen Hx. This means the leak is downstream of the RHX in containment. Since Charging flow increased from a normal value of 132 GPM to 137 GPM with no other changes the leak rate is approximately 5 GPM and will required isolation of the charging header to isolate the leak.

C - INCORRECT. Leak is in the charging header. This leakage would cause a rise in letdown flow through the Regen HX and a rise on 2-QTA-160, Regen HX Outlet Temp - Letdown.

D - INCORRECT. Wrong location for leak. This leakage would cause a rise on 2-QTA-160, Regen HX Outlet Temp - Letdown due to less Charging Flow to cool the letdown exiting the regen HX.

LESSON PLAN/OBJ: RO-C-AOP-D1/#RO-C-AOP0160412-E1
REFERENCE: SOD-00300-001

KA - 004000 2.4.47
Chemical and Volume Control System (CVCS)
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

KA Justification - Requires use of reference material (instrument readings and drawing) to determine the location RCS leakage on in the CVCS system.

Original Question # - RO26 AUDIT-4, SEQ2007
Original Question KA - 004000 K6.07

22. 022 004/BOTH/OK/MODIFIED/NRC EXAM 2007-7/004000 K3.08/3.6/3.8/H/3

Given the following conditions:

- QRV-200, RCP Seal Backpressure Valve, is operating at 30% open.

Assuming QRV-251, Charging Line Flow Control Valve is NOT adjusted, IF QRV-200 fails to 60% open, THEN:

	Charging Pump Discharge Press	**RCP Seal Injection Flow**	**Charging Flow to Regen Hx**
A.	Lowers	Rises	Lowers
B.	Rises	Lowers	Rises
C.	Rises	Rises	Lowers
D.✓	Lowers	Lowers	Rises

ANSWER: D

A - INCORRECT. Seal injection flow lowers and Charging Flow rises. Plausible due to relationship between RCP Seal Injection flow and Charging is correct but the directions are wrong. Student must know how the valve failures impact system operation.

B - INCORRECT. CCP discharge pressure lowers and Charging Flow rises. Plausible due to relationship between RCP Seal Injection flow and Charging is correct but the impact on Charging pump discharge pressure is wrong. Student must know how the valve failures impact system operation.

C - INCORRECT. CCP discharge pressure lowers, Seal injection flow lowers, and Charging Flow rises. Plausible due to relationship between RCP Seal Injection flow and Charging is correct but the directions are wrong but the impact on Charging pump discharge pressure is wrong. Student must know how the valve failures impact system operation.

D - CORRECT. QRV-200 will cause a lower backpressure on the CCP discharge and seal injection line, resulting in lower CCP discharge pressure and less flow to the RCP seals. In addition this action will raise charging flow.

LESSON PLAN/OBJ: RO-C-00300/#3
REFERENCE: SOD-00300-001

MODIFIED: Changed stem to go from 30% open to 60% open. Changed the correct answer to "D"

KA - 004000 K3.08
Chemical and Volume Control System (CVCS)
Knowledge of the effect that a loss or malfunction of the CVCS will have on the following:
RCP seal injection
RO - 3.6 SRO - 3.8
CFR - 41.7 / 45.6

KA Justification - Requires the knowledge of a malfunction of a CVCS component (QRV-251) will have on RCP Seal Injection.

Original Question # - INPO # 28845 Indian Point Unit 2 - 12/9/2004, NRC EXAM 2007-7
Original Question KA - 000022 AK1.02

23. 023 002/BOTH/OK/DIRECT/NRC EXAM 2006-029-45/005000 K6.03/2.5/2.6/H/3

Given the following conditions:

- Unit 2 is in Mode 4 during cooldown per 2-OHP-4021-001-004, Plant Cooldown from Hot Standby to Cold Shutdown
- West RHR Pump and Heat Exchanger are operating, aligned to the cooldown path through injection lines to Cold Legs Loops 2 & 3
- RCS temperature is 300°F and stable
- RCS pressure is 335 psig and stable

The air supply line to IRV-320, West RHR Hx Outlet Valve, breaks, causing a complete loss of Instrument Air to the valve.

Which ONE of the following describes the effect on the plant and the action that could be taken to mitigate the transient?

A. RHR Flow through the West HX will be lost. Throttle open IRV-311 RHR HX Bypass to maintain greater than 3000 gpm RHR flow.

B. RHR Flow through the West HX will be lost. Stop the West RHR pump immediately to prevent overpressurizing letdown.

C. RHR Flow through the West HX will rise. Throttle ICM-111, RHR Discharge to Cold Leg 2 & 3 and IRV-311 RHR HX Bypass to prevent overcooling the RCS.

D✓ RHR Flow through the West HX will rise. Throttle ICM-321, West RHR Injection to Loops 2 & 3 and IRV-311 RHR HX Bypass to prevent overcooling the RCS.

ANSWER: D

A - INCORRECT. IRV-320 fails open so flow will raise. Plausible since RHR flow is maintained > 3000 gpm to minimize vibrations & cavitation through the piping.

B - INCORRECT. IRV-320 fails open so flow will raise. Plausible since RHR flow to the letdown system taps off before the IRV-320 and if IRV-320 closed it may raise letdown pressure.

C - INCORRECT. The ICM-111, RHR Discharge to Cold Leg 2 & 3 is in the NORMAL Cooldown Path This would be correct if the Normal Cooldown path was in service. (2-OHP-4021-017-002, Step 4.13)

D - CORRECT. IRV-320 fails open on loss of air. This will raise RHR flow through the HX. ICM-321 can be throttled closed to reduce total RHR flow and IRV-311 can be throttled open to allow more flow to bypass the HX in order to control RCS cooldown.

LESSON PLAN/OBJ: RO-C-01700/#RO-C-01700-E4, RO-C-01700-E6
REFERENCE: 2-OHP-4021-017-002, Placing in Service the RHR System; 2-OHP-4022-064-002, Loss of Control Air Recovery (Step 43 & Att. B-9), SOD-01700-001

KA - 005000 K6.03
Residual Heat Removal System (RHRS)
Knowledge of the effect of a loss or malfunction of the following will have on the RHRS:
RHR heat exchanger
RO - 2.5 SRO - 2.6
CFR - 41.7 / 45.7

KA Justification - Requires knowledge of how to control RCS cooldown rate following a malfunction of the RHR Hx Out valve full open.

Original Question # - Cook 2006 NRC Exam - 029-45
Original Question KA - 005000 A2.01

24. 024 003/BOTH/OK/DIRECT/NRC EXAM 2004-069-5/006000 K1.11/2.8/3.2/F/3

Given the following conditions:

- Unit 2 has experienced a loss of both CCW pumps in MODE 3
- NEITHER Unit 2 CCW pump can be restarted.
- CVCS crosstie from Unit 1 is NOT available.
- BOTH Unit 2 CCPs are running because a CCP swap was in progress.
- 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 both CCPs.

B. Immediately stop one CCP; stop the second CCP within 1-1/2 minutes of the event.

C. Stop BOTH CCPs within 1-1/2 minutes of the event.

D.✓ Immediately stop one CCP; run the second CCP as long as it continues to operate.

ANSWER: D

A - INCORRECT. One pump should be run as long as possible to allow time to align Seal injection crosstie. Plausible due to knowledge that the CCP's will fail in a short time frame without cooling.

B - INCORRECT. One pump should be run as long as possible to allow time to align Seal injection crosstie. (The pump may trip after 1.5 minutes) Plausible due to knowledge that the CCP's will fail in a short time frame without cooling and the allowable time to operate with no cooling is 90 seconds. This saves one pump for a later period.

C - INCORRECT. One pump should be run as long as possible to allow time to align Seal injection crosstie. (The pump may trip after 1.5 minutes) Plausible due to knowledge that the CCP's will fail in a short time frame without cooling and the allowable time to operate with no cooling is 90 seconds. This saves both pumps for a later period.

D - 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 Seal injection crosstie.

LESSON PLAN/OBJ: RO-C-AOP-D14\#RO-C-AOP0560412-E3
REFERENCE: 2-OHP-4022-016-004, Loss of Component Cooling Water

KA - 006000 K1.11
Emergency Core Cooling System (ECCS)
Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems:
CCWS
RO - 2.8 SRO - 3.2
CFR - 41.2 to 41.9 / 45.7 to 45.8

KA Justification - Requires knowledge of the cause and effect relationship between the a loss of CCW to the CCP (ECCS Pump) and the ability of the CCP to continue to operate.

Original Question # - Master AOP1CAOP5.13, NRC Exam 2004-069-5
Original Question KA - Unknown

25. 025 004/BOTH/OK/NEW/NEW/007000 A4.10/3.6/3.8/F/3

Given the following conditions:

- The Plant has just completed a Heatup to Normal Operating Temperature and Pressure.
- Operators suspect a small leak through Pressurizer Safety Valve SV-45B.

What indication combinations are available to help the operator determine if this valve is faulted?

A. A significant PRT Temperature Rise (>200ºF)
The Common Safety Valve Tailpipe Temperature indicator
The Common Safety Valve line acoustic monitor

B. A significant PRT Temperature Rise (>200ºF)
The Safety Valve SV-45B Tailpipe Temperature indicator
The Safety Valve SV-45B line acoustic monitor

C. A slight PRT Temperature Rise (<50ºF)
The Safety Valve SV-45B Tailpipe Temperature indicator
The Common Safety Valve line acoustic monitor

D✓ A slight PRT Temperature Rise (<50ºF)
The Safety Valve SV-45B Tailpipe Temperature indicator
The Safety Valve SV-45B line acoustic monitor

ANSWER: D

A - INCORRECT. A large temperature indication of > 200ºF would only be expected if the safety was failed mostly open. PORVs share indicators and safeties have separate indicators. Plausible but due to constant enthalpy from throttling process the temperature change can not be 200ºF as well as the Safeties have individual temp and acoustic monitors unlike the PORV's with a common set of temp and acoustic monitors.

B - INCORRECT. A large temperature indication of > 200ºF would only be expected if the safety was failed mostly open. It is correct that the safeties have separate indicators. – Plausible but due to constant enthalpy from throttling process the temperature change can not be 200ºF the remaining portion of the answer is correct.

C - INCORRECT. PORVs share indicators and safeties have separate indicators for the acoustic line. Plausible due to correct anticipated change in temperature but the remaining portion of the answer is not correct for the Safeties – the PORV's have the common set of temperature and acoustic monitors.

D - CORRECT. For a small leak into the PRT there should not be a significant rise in PRT temperature. PORVs share indicators and safeties have separate indicators for the acoustic line.

LESSON PLAN/OBJ: RO-C-00202/#4, RO-C-EOP09/#22
REFERENCE: RO-C-00202 pg. 41

KA - 007000 A4.10
Pressurizer Relief Tank/Quench Tank System (PRTS)
Ability to manually operate and/or monitor in the control room:
Recognition of leaking PORV/code safety
RO - 3.6 SRO - 3.8
CFR - 41.7 / 45.5 to 45.8

KA Justification - Question tests the ability of the operator to monitor (by identifying expected PRT Temperature trend and available indications) the PRT Temperature and associated connections (PORV/SAFETY lines) to help determine which Safety is leaking

Original Question # - New
Original Question KA - New

26. 026 004/BOTH/OK/MODIFIED/NRC EXAM 2007-40/013000 K5.02/2.9/3.3/H/4

Given the following conditions:

- Containment pressure instrument Channel #1, 2-PPP-303, declared inoperable.
- Required actions per 2-OHP-4022-013-011, Containment Instrumentation Malfunction, have been completed.
- Required Technical Specification Actions have been taken for Channel #1, 2-PPP-303.

Which ONE of the following describes the SI and CTS, and Containment Isolation Phase A (CIA) and B (CIB) response to a subsequent failure of CRID 4 power supply.

	SI ACTUATES	CTS ACTUATES	CIA ACTUATES	CIB ACTUATES
A.	YES	NO	YES	NO
B.	YES	YES	YES	YES
C.	NO	YES	NO	YES
D.✓	NO	NO	NO	NO

ANSWER: D

A - INCORRECT. See explanation Below.
B - INCORRECT. See explanation Below.
C - INCORRECT. See explanation Below.
D - CORRECT. See explanation Below.

The CTS Actuation Bistable is placed in the BYPASSED condition to prevent inadvertent actuation. This changes the remaining channel coincidence to 2/3 instead of the previous 2/4. Only 3 channels (Channels 2, 3, & 4) feed the SI Actuation. The bistable for the CTS actuation is placed in the BYPASS condition, making the CTS a 2/3 coincidence for the remaining channels (2, 3, and 4). CRID 4 failure will NOT meet the 2/3 co-incidence for either the SI or CIA. CTS/CIB still required 2/3 to actuate, therefore only one channel will not cause the CTS/CIB. The logic between CTS and CIB is always the same which prevents the student from eliminating distractors based on obvious distractors; this same logic applies to SI and CIA. This forces the student to have a full understanding of the question in order to answer the question correctly.

LESSON PLAN/OBJ: RO-C-01100/#6
REFERENCE: 2-OHP-4022-013-011 Containment Instrumentation Malfunction

MODIFIED: Changed from Channel #4 to Channel #1 (PPP-303). Changed to a CRID 4 failure. Changed correct answer to "D."

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

KA Justification - Requires the knowledge of the logic coincidence for both the SI/CIA and CTS/CIB functions and how the redundancy of instruments allows for single failure and will still actuate as required for multiple failures (reliability).

Original Question # - COOK04-037, NRC EXAM 2007-40
Original Question KA - 013000 K2.01

27. 027 001/BOTH/OK/NEW/NEW/008000 A2.05/3.3/3.5/H/3

Given the following conditions:

- Unit 1 and Unit 2 are operating at 100% Power.
- North Spent Fuel Pit pump and cooler in service
- Spent Fuel Pit at Normal Level
- Spent Fuel Pit contains 2015 spent fuel assemblies
- Refueling Water Purification aligned to purify Unit 1 RWST

The 1-CRV-445, CCW from North SFP Hx, control air supply line ruptures causing valve to fail.

Which ONE of the following conditions describes the impact on the Spent Fuel Pool Cooling system and the actions needed to address this condition?

A.✓ SFP Temperature will rise.
Place the South SFP Pump and Hx in service.

B. SFP Temperature will rise.
Manually control SFP temperature using 1-CRV-445 bypass valve.

C. SFP Temperature will lower.
Place the South SFP Pump and Hx in service.

D. SFP Temperature will lower.
Isolate 1-CRV-445 and manually control SFP temperature using 1-CRV-445 bypass valve.

ANSWER: A

A - CORRECT. 1-CRV-445 fails closed on loss of air. Due to the heat load in the SFP, the SFP temperature will rise. Impacts the ability of Unit 1 CCW system to provide SFP Cooling. Unit 2 SFP cooling loop will need to be placed in service.

B - INCORRECT. 1-CRV-445 fails closed on loss of air. Due to the heat load in the SFP, the SFP temperature will rise. 1-CRV-445 does not have a bypass valve however many air operated valves in the plant do have bypass valves around them which makes this a valid distractor.

C - INCORRECT. Temperature would lower if 1-CRV-445 failed open on loss of air.

D - INCORRECT. Temperature would lower if 1-CRV-445 failed open on loss of air. 1-CRV-445 does not have a bypass valve however many air operated valves in the plant do have bypass valves around them which makes this a valid distractor

LESSON PLAN/OBJ: RO-C-01600/#RO-C-01600-E6
REFERENCE: OP-1-5135B, SOD-01600-001, COMPONENT COOLING WATER SYSTEM

KA - 008000 A2.05
Component Cooling Water System (CCWS)
Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Effect of loss of instrument and control air on the position of the CCW valves that are air operated
RO - 3.3 SRO - 3.5
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - requires the ability to predict the impact of loss of air to 1-CRV-445 and the ability of the CCW system to cool the SFP. Based on this impact, requires knowledge of the actions required to control the consequences the malfunction.

Original Question # - New
Original Question KA - New

28. 028 003/BOTH/OK/MODIFIED/NRC EXAM 2008-23/00WE03 EA1.1/4.0/4.0/H/3

Given the following conditions on Unit 1:

- A Small Break LOCA has occurred.
- RCS Wide Range Pressure lowered to 1350 psig and is stable.
- Containment pressure has remained less than 2.8 psig.
- 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 depressurize the RCS.

B. All Pressurizer PORVs to depressurize the RCS.

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

D✓ PZR Normal Spray Control valve(s) to spray down the PZR steam space.

ANSWER: D

A. INCORRECT. Since normal spray are available (RCS Wide Range Pressure above the RCP Trip Criteria), sprays would be used before one PZR PORV.
B. INCORRECT. Opening MORE THAN ONE PORV is NOT an appropriate action. If this option were used, then only one PORV would be used to minimize the potential for a PORV sticking open.
C. INCORRECT. This action is a third option in the event that a PORV is not available.
D - CORRECT. This is the "normal" method used to depressurize the RCS in ES-1.2.

Note: **ALL distractors are valid methods that can be used to depressurize the RC and accomplish the intended goal of refilling the pressurizer. The student must be able to determine which action is the most correct based on plant conditions.**

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

MODIFIED: Removed loss of offsite power and replaced with RCS Wide Range Pressure above the RCP Trip criteria from the Foldout Page. Changed correct answer to "D."

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

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.

Original Question # - INPO Bank #30436 - KEWAUNEE-222006, NRC EXAM 2008-23
Original Question KA - WE03EA1.1

29. 029 003/BOTH/OK/DIRECT/RO26 AUDIT-61/00WE04 EK2.1/3.5/3.9/H/3

The plant was in Mode 1. Reactor trip and safety injection have occurred. Due to high Aux Building radiation levels, the crew has entered 2-OHP-4023-ECA-1.2, LOCA Outside Containment. Actions have been taken in an attempt to isolate the break.

Given the following plant conditions:
- PZR level is off-scale low
- SI pump flow is 0 GPM
- RCS pressure is 1700 psig and rising.
- Aux Building Radiation Monitors are in alarm

Which ONE of the following describes the status of the leak based on the requirements of 2-OHP-4023-ECA-1.2?

A. The leak is isolated based on SI flow of 0 GPM

B.✓ The leak is isolated based on RCS pressure rising.

C. The leak is NOT isolated based on PZR level indication not rising.

D. The leak is NOT isolated based on Aux Building radiation monitor indication.

ANSWER: B

A - INCORRECT.	SI flow would be 0 if RCS pressure is above shutoff head of the SI Pump. Plausible due to not having any injection flow could mean that the leak is isolated but this is not what is directed by the procedure
B - CORRECT.	RCS pressure is the required parameter for determination of isolation Incorrect.
C - INCORRECT.	PZR level is not used, but it will rise after awhile when RCS inventory is restored. Plausible due to Fold out page in several EOP's require re-initiation of SI if PZR level is at described level in stem of question.
D - INCORRECT.	Plausible since Aux Building radiation is used as an entry condition to the procedure.

LESSON PLAN/OBJ: RO-C-EOP09/#34
REFERENCE: 2-OHP-4023-ECA-1.2, LOCA Outside Containment,
RO-C-EOP09 pg. 252

KA - 00WE04 EK2.1
LOCA Outside Containment
Knowledge of the interrelations between the LOCA Outside Containment and the following:
Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features
RO - 3.5 SRO - 3.9
CFR - 41.7 / 45.7

KA Justification - Question Addresses a LOCA Outside Containment and how the plant & instrumentation responds (interrelations) based on closure of valves (operation of components).

Original Question # - RO26 AUDIT-61, GINNA2007

30. 030 005/BOTH/OK/DIRECT/MASTER 01EOPC1110-7/00WE05 EK3.3/4.0/4.1/F/3

The control room operators are responding to a red path on the Heat Sink CSF. While attempting to restore feed flow to a SG in accordance with OHP-4023-FR-H.1, Response to Loss of Secondary Heat Sink, conditions degrade to the point that RCS bleed-and-feed must be established.

Under these conditions, RCS bleed-and-feed must be established expeditiously to:

A. prevent a loss of secondary heat sink.

B.✓ minimize core uncovery and prevent inadequate core cooling.

C. prevent an overpressurization challenge to the reactor vessel.

D. prevent a rapid RCS overpressurization, followed by a rapid RCS depressurization due to RCP seal failure.

ANSWER: B

A - INCORRECT. Attempts to restore the heat sink have been unsuccessful. Bleed-and-feed is established to raise the amount of injection flow into the core and thus minimize the core uncovery. Plausible based on the purpose of the FRP being used. The goal is to establish and thus prevent a loss of secondary heat sink.

B - CORRECT. If the operator cannot restore feedwater flow to the SGs, conditions will degrade to the point where RCS bleed and feed must be established to minimize core uncovery and prevent inadequate core cooling.

C - INCORRECT. Based on the FRGs, even if overpressurization were a concern, the priority of core cooling is higher than PTS concerns. Plausible based on the event if no actions were taken the RCS would increase in pressure based on the lack of heat removal from the RCS.

D - INCORRECT. Core cooling is the second highest priority in the EOPs (Subcriticality being the highest). Any other concern would be of a lower priority than establishing core cooling of some nature, in this case bleed-and-feed. Plausible based on the conditions described do not state the basis for requiring feed and bleed (the conditions could be caused by high RCS pressure) and the conditions could also exist without any Charging pumps running thus challenging the RCP seals.

LESSON PLAN/OBJ: RO-C-EOP11/#10
REFERENCE: RO-C-EOP11 Study Guide, FR-H.1 Background

KA - 00WE05 EK3.3
Loss of Secondary Heat Sink
Knowledge of the reasons for the following responses as they apply to the Loss of Secondary Heat Sink:
Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations
RO - 4.0 SRO - 4.1
CFR - 41.5 / 41.10 / 45.6 / 45.13

KA Justification - Requires knowledge of the reason for implementing bleed-and-feed in the EOPs during a Loss of Heat Sink event when heat sink cannot be restored.

Original Question # - 01EOPC1110-7
Original Question KA - EPE 005 EK3.1

31. 031 003/BOTH/OK/DIRECT/RO24 AUDIT-023-7/W/E09 EK3.2/3.2/3.6/F/3

Given the following conditions:

- Unit 2 Reactor Tripped due to a loss of offsite power
- The crew is implementing 2-OHP-4023-ES-0.2, Natural Circulation Cooldown

Which ONE fo the following describes the reason for maintaining subcooling greater than 90°F if ALL CRD fans are running OR greater than 220°F if less than ALL CRD fans are running during the cooldown.

A. To collapse any voids formed in the CRD housings.

B.✔ To prevent possible void formation in the upper head.

C. To prevent degradation of reactor coolant pump seals due to steam.

D. To ensure adequate subcooling due to possible degradation of core exit T/Cs accuracy.

ANSWER: B

A - INCORRECT. Plausible since loss of CRD fans would lead to high temperatures in CRD housing cooling, but this is not the reason for requiring higher subcooling.

B - CORRECT. 2-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 90°F.

C - INCORRECT. Plausible since overheating and steam formation in the RCP seals is a concern on loss of all AC.

D - INCORRECT. Plausible since TCs are compensated for inaccuracies, but this is not the reason for requiring higher subcooling.

LESSON PLAN/OBJ: RO-C-EOP03/#12
REFERENCE: 2-OHP-4023-ES-0-2, Natural Circulation Cooldown, step 14

KA - 00WE09 EK3.2
Natural Circulation Operations
Knowledge of the reasons for the following responses as they apply to the Natural Circulation Operations:
Normal, abnormal and emergency operating procedures associated with Natural Circulation Operations
RO - 3.2 SRO - 3.6
CFR - 41.5 / 41.10 / 45.6 / 45.13

KA Justification - Requires knowledge of the reason for the requirement to maintain adequate subcooling margin during a natural circ cooldown event.

Original Question # - RO24 AUDIT-023-7
Original Question KA - W/E09 EK2.1

32. 032 003/BOTH/OK/DIRECT/CATAWBA2005/00WE11 EA2.2/3.4/4.2/F/3

Given the following conditions:

- 1-OHP-4023-ECA-1.1, Loss of Emergency Coolant Recirculation, has just been entered
- Refueling Water Storage Tank (RWST) level is 5.5%

Which ONE of the following procedure actions is performed FIRST?

A. Initiate makeup to the U-1 RWST from the Boric Acid Blender.

B. Start one reactor coolant pump.

C. Initiate makeup to the U-1 RWST from the U-2 RWST.

D.✓ Secure all ECCS and CTS pumps taking a suction from the RWST.

ANSWER: D

A - INCORRECT. Makeup is started in step 7. Plausible as this is a step in the procedure but does not occur prior to the requirement off the fold out page.

B - INCORRECT. RCPs are not started until later. Plausible as the procedure does start RCP's but does not occur prior to the requirement off the fold out page.

C - INCORRECT. Makeup from the Opposite unit RWST is not used until step 7. Plausible as the procedure does accomplish this action but not until after the fold out page has been implemented .

D - CORRECT. The Foldout Page has actions to secure RHR & CTS when level is < 11% and CCP & SI pumps when level is <7%. This would be the first action taken upon entering the procedure.

LESSON PLAN/OBJ: RO-C-EOP09\#35
REFERENCE: 1-OHP-4023-ECA-1.1 Foldout Page. (Note that this action is also listed as a critical task for ECA-1.1 RO-C-EOP09 pg. 106)

KA - 00WE11 EA2.2
Loss of Emergency Coolant Recirculation
Ability to determine and interpret the following as they apply to the Loss of Emergency Coolant Recirculation:
Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments
RO - 3.4 SRO - 4.2
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Requires the Ability to determine and interpret procedural requirements for stopping ECCS and CTS pumps on low RWST level to keep plant operating within the limitations in the facility's license and amendments.

Original Question # - RO25 AUDIT-17, CATAWBA2005
Original Question KA - 00WE11 EA2.2 3.4/4.2 CFR 43.5/45.13

33. 033 002/BOTH/OK/DIRECT/SEQ2007/00WE12 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:

- Containment pressure is 3 psig.
- The crew has taken action to minimize the plant cooldown.
- Steam Generator AFW flow indicates 25x10^3 pph each SG.
- T-hots are slowly lowering.
- The following alarms are received:
  - Ann. 213 Drop 5, STEAM GEN #1 WATER LEVEL LOW-LOW
  - Ann. 213 Drop 35, STEAM GEN #2 WATER LEVEL LOW-LOW
  - Ann. 214 Drop 5, STEAM GEN #3 WATER LEVEL LOW-LOW
  - Ann. 214 Drop 35, STEAM GEN #4 WATER LEVEL LOW-LOW

Which ONE of the following actions is required in accordance with 2-OHP-4023-ECA-2.1?

A. Adjust AFW flow to 60x10^3 pph on each Steam Generator. The minimum NR level, per 2-OHP-4023-ECA-2.1, is 28%.

B. Adjust AFW flow to 60x10^3 pph on each Steam Generator. The minimum NR level , per 2-OHP-4023-ECA-2.1 , is 50%.

C.✓ Maintain AFW flow at its current value. If T-hot starts to rise, raise AFW flow to stabilize RCS temperature.

D. Maintain AFW flow at its current value. If SG levels continue to lower, raise AFW flow to maintain SG levels >13% to prevent a transition to 2-OHP-4023-FR-H.1, Response to Loss of Secondary Heat Sink.

ANSWER: C

A. INCORRECT. After throttling to minimize RCS cooldown, even if levels are low, AFW remains throttled until Thot begins to rise. At that point, AFW is throttled just enough to stabilize temperature. Credible because 28% is the lower limit that level is maintained for Adverse Containment. (Containment is NOT Adverse)

B. INCORRECT. After throttling to minimize RCS cooldown, even if levels are low, AFW remains throttled until Thot begins to rise. At that point, AFW is throttled just enough to stabilize temperature. Credible because 50% is the upper limit that level in most of the EOPs.

C. CORRECT. AFW Flow is maintained at a minimum amount since the levels are low and T-hot is lowering.

D. INCORRECT. Flow is maintained per at $25x10^3$ pph per the procedure. If the transition to FR H.1 is reached Step 1 will return the Crew to ECA-2.1. Plausible as Operator should know they have the ability to avoid a Red path on heat sink but avoiding this red path is in violation of ECA 2.1 requirements and the red path addresses the intentional reduction in AFW flow.

Note:

- **Level setpoint for SG Low-Low is 22% - TDAFP start setpoint.**
- **Since this an operator induced reduction of AFW flow, FR-H. 1 actions would not be performed even if the transition was made.**

LESSON PLAN/OBJ: RO-C-05100\#12, RO-C-EOP07/#8
REFERENCE: 2-OHP-4024-213 & 214 Drops 5 & 35, 2-OHP-4023-ECA-2.1, RO-C-EOP07

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

KA Justification - Requires knowledge of the operational implications of the throttling AFW flow to the SGs during an Uncontrolled Depressurization of all Steam Generators and the provisions for when to raise flow rates.

Original Question # - RO26 AUDIT-64, SEQ2007
Original Question KA - 00WE12 2.4.31

34. 034 002/BOTH/OK/MODIFIED/NRC EXAM 2008-21/00WE14 EK2.2/3.4/3.8/H/3

Given the following conditions on Unit 2:

- A LOCA occurred 60 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 if the CTS pump trips.

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

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

ANSWER: A

A - CORRECT. RHR has injected for 50 minutes. (A LOCA occurred on Unit 2 sixty minutes ago.)

B - INCORRECT. 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. Plausible since the student should know some form of containment pressure suppression is required that action would be required if the only running CTS pump trips. Distractor requires the student to know the requirement for pressure and time since accident to determine the correct answer.

C - INCORRECT. RHR spray is required 50 minutes after the accident. It is assumed that RHR will be on Recirculation at this time. Plausible as concern for available NPSH to the running RHR pump is a concern for accident mitigation. Distractor requires the student to know the water level in containment supports the NPSH requirements for establishing RHR spray.

D - INCORRECT. After RHR has injected for 50 minutes the core is sufficiently cooled to allow RHR to be diverted to support spray functions. Plausible due to concern for placing the running RHR pump in a runout condition from the increase in flow thru the pump.

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

Modified: Changed stem to 60 minutes which makes A correct Answer. Changed Distractor B (former correct answer) from wait until 50 minutes to ONLY required if CTS trips. - SWP 1-21-10

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

KA Justification - Questions tests the knowledge of how and when the heat removal system is used to aid in controlling a high containment pressure.

Original Question # - Cook NRC Exam 2002-026-1, 01EOPC1313-2, NRC EXAM 2008-21
Original Question KA - 00WE14 EK2.2

35. 035 005/BOTH/OK/DIRECT/NRC EXAM 2004-074-5/00WE16 EA2.2/3.0/3.3/F/4

Chemistry had confirmed two leaking fuel rods on Unit 1 when a Small Break LOCA occurred 12 hours ago.

The following conditions exist on Unit 1:
- All Red and Orange Paths have been addressed.
- Containment pressure is 1.0 psig.
- Containment air temperature is 215ºF.
- Lower Containment high range area monitors, (VRA-1310/1410) are reading 10 R/HR
- 1-OHP-4023-FR-Z.3, Response to High Containment Radiation Level, is entered.

In accordance with 1-OHP-4023-FR-Z.3, which ONE of the following must be verified?

A. Both Containment Recirculation Fans (CEQ) are running.

B. Upper and Lower Containment Ventilation Fans (CUV/CLV) are running.

C✓ Containment Ventilation Isolation has occurred.

D. Control Room Ventilation System is in ISOLATE.

ANSWER: C

A - INCORRECT. Containment Recirculation Fans are run to help reduce Hydrogen Buildup. They are NOT run in 1-OHP-4023-FR-Z.3.

B - INCORRECT. Containment Ventilation fans are tripped on a Containment Isolation signal. Plausible as circulating the containment atmosphere is a logical action to aid in the removal of the radiation.

C - CORRECT. 1-OHP-4023-FR-Z.3 requires the crew to verify Containment Ventilation Isolation.

D - INCORRECT. Control Room Ventilation is aligned during a SI but is not addressed in 1-OHP-4023-FR-Z.3. Plausible as it is a required action to ensure the Control Room staff limits accident dose rates but this action was previously completed and is not addressed in Z.3.

LESSON PLAN/OBJ: RO-C-EOP13/#6
REFERENCE: 1-OHP-4023-FR-Z.3, Response to High Containment Radiation Level pg. 2

KA - 00WE16 EA2.2
High Containment Radiation
Ability to determine and interpret the following as they apply to the High Containment Radiation:
Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments
RO - 3.0 SRO - 3.3
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Requires the ability to determine the major action category (required actions) associated with 1-OHP-4023-FR-Z.3, Response to High Containment Radiation Level.

Original Question # - NRC EXAM 2004-074-5
Original Question KA - 000061 2.4.6

36. 036 004/BOTH/OK/MODIFIED/RO25 AUDIT-34/010000 A2.01/3.3/3.6/H/3

Given the following conditions:

- Unit 1 in Mode 4 cooling down to Mode 5
- Pressurizer level is 80%
- 11PHC Pressurizer heater groups are in MANUAL and ON
- Reactor coolant pumps (RCP) #13 and #14 are running
- An electrical fault results in the loss of RCP Bus 1D and T11D

Five minutes later it is reported the pressurizer outflow cannot be verified.

Which of the following actions will reinitiate and then maintain a continuous pressurizer outflow?

A. Verify pressurizer heaters from 11PHC output current and close NRV-164 Loop 4 PZR Spray Control valve

B.✓ Energize pressurizer heaters from 11PHA and close NRV-163 Loop 3 PZR Spray Control valve

C. Raise the demand on NRV-164 Loop 4 PZR Spray Control valve

D. Adjust charging and letdown to raise pressurizer level to 85%

ANSWER: B

A - INCORRECT. The loss of T11D causes the loss of PZR Heaters 11PHC.

B - CORRECT. The loss of RCP Bus 1D causes the loss of RCP #3. The loss of T11D causes the loss of PZR Heaters 11PHC. The operator should energize the 11PHA heaters and close the PZR Spray valve associated with the tripped RCP.

C - INCORRECT. The loss of T11D causes the loss of PZR Heaters 11PHC. Raise the demand on the Spray valve will lower pressure and cause inflow.

D - INCORRECT. Raising level will cause an inflow.

LESSON PLAN/OBJ: RO-C-00202\#14, RO-C-NOP2\#RO-C-NOP2-E13
REFERENCE: RO-C-NOP2, OHP-4021-001-004 pg. 20, SOD-00202-002

**MODIFIED: Changed Heater group energized, & loss of Power to RCP #14.
Modified distractors to make B correct & Changed A.**

KA - 010000 A2.01
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:
Heater failures
RO - 3.3 SRO - 3.6
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Question requires operator to predict response (determine status of components available) of the PZR pressure control system due to a loss of power (and heaters) and use procedures/actions to correct the conditions resulting from the failure.

Original Question # - RO25 AUDIT-34, modified from CATAWBA2005
Original Question KA - SYS010 A2.01

37. 037 004/BOTH/OK/NEW/NEW/011000 K5.05/2.8/3.1/H/4

Unit 2 was operating at 100% power when an instrument failure caused Pressurizer Level to rise to 73%.

The Crew has restored the level control instrumentation and has stabilized charging and letdown
The following conditions exist at 1300 hours:

- 2-QRV-160 and 2-QRV-161, Letdown Orifice Valves are Open
- Letdown Hx Outlet Flow QFI -301 118 gpm
- Charging Header Flow QFI -200 120 gpm
- Total seal flow to RCPs QFI -210 to 240 32 gpm
- Pressurizer Level NLP-151 to 153 73%

Assuming no operator actions, design RCP seal return flows, and pressurizer volume of 75 gallons/% at what time will the Pressurizer Level be returned to the 100% program value?

A. 1348 hrs.

B. 1406 hrs.

C.✓ 1522 hrs.

D. 1618 hrs.

ANSWER: C

A - INCORRECT. Time is based on time to Unit 2 Pressurizer level of 54.1% assuming 30 gpm mismatch (Letdown + Seal Inj - Charging) = 47.25 minutes

B - INCORRECT. Time is based on time to Unit 1 Pressurizer level of 46.6% assuming 30 gpm mismatch (Letdown + Seal Inj - Charging) = 66 minutes

C - CORRECT. Unit 2 100% PZR level is 54.1%. With charging flow at 120 gpm and letdown at 118 with 12 gpm from the seals, a net of 10 gpm is being removed from the RCS system. Based on this and a conversion of ~75 gallons/% level in the PZR (either unit), level will reach the program level setpoint of 54.1% at 1522 hrs (T+141.75 minutes - 18.9%=1417.5 gallons).

D - INCORRECT. Time is based on Unit 1 Initial Pressurizer Level of 46.6% with 10 GPM mismatch = 198 minutes

Note: Pressurizer volume based on 16.6 ft^3/% x 1lbm/.0267 ft^3 x 1gal/8.35lbm

LESSON PLAN/OBJ: RO-C-00202/#RO-C-00202-E12
REFERENCE: SOD-00202-003, SOD-00300-001

KA - 011000 K5.05
Pressurizer Level Control System (PZR LCS)
Knowledge of the operational implications of the following concepts as they apply to the PZR LCS:
Interrelation of indicated charging flow rate with volume of water required to bring PZR level back to programmed level hot/cold
RO - 2.8 SRO - 3.1
CFR - 41.5 / 45.7

KA Justification - Question tests the operational knowledge of how long it will take to restore the pressurizer level to program based on net charging flow.

Original Question # - New
Original Question KA - New

38. 038 005/BOTH/OK/DIRECT/NRC EXAM 2004-103-4/012000 A3.04/2.8/2.9/H/3

Given the following conditions:

- Pressurizer Pressure Channel #1 has failed and has been placed in the tripped condition.
- Reactor trip breaker testing was taking place at 75% power.
- Pressurizer Pressure Channel #2 has spiked low causing an inadvertent Safety Injection Actuation and a reactor trip on Unit 1.
- Pressurizer Pressure Channel #2 has returned to a normal reading.

The following conditions currently exist:

- Reactor trip breaker A: **OPEN**
- Reactor trip bypass breaker A: **OPEN**
- Reactor trip breaker B: **OPEN**
- Reactor trip bypass breaker B: **CLOSED**

Which ONE of the following describes the impact (if any) this condition will have on restoring the plant to stable conditions?

A. The Train B Safety Injection signal will NOT be able to be reset. Train B equipment will have to be placed in Pull-to-Lockout to stop it.

B✓ The Train B Safety Injection signal will reset but Auto Safety Injection Actuation will NOT be blocked.

C. The Safety Injection signal will NOT be able to be reset on either train. Safeguards equipment will have to be placed in Pull-to-Lockout to stop it.

D. The Safety Injection signal will reset on both trains. Auto Safety Injection Actuation will be blocked.

ANSWER: B

A - INCORRECT. The Safety Injection signal will reset.
B - CORRECT. The SI reset and P-4 block features are train specific. With a failure of Train B reactor Trip Bypass Breaker to open a P-4 signal is not generated on Train B. Since the cause of the SI was a pressure channel spike the SI signal is NOT preventing Train B from being reset. The SI will reset but the Auto SI blocking function of P-4 will NOT function on Train B.
C - INCORRECT. The Safety Injection signal will reset. The SI reset and P-4 block features are train specific.
D - INCORRECT. The SI reset and P-4 block features are train specific so Train B auto SI will NOT be blocked.

LESSON PLAN/OBJ: RO-C-01100 / #6
REFERENCE: OP-2-98512-21 Safeguard actuation & Reactor Trip Signals Logic Diagram

KA - 012000 A3.04
Reactor Protection System
Ability to monitor automatic operation of the RPS, including:
Circuit breaker
RO - 2.8 SRO - 2.9
CFR - 41.7 / 45.5

KA Justification - Question tests ability of operator to determine if auto action (SI) has occurred and the impact that a circuit breaker (RTB) will have on this action/RPS system. The Operator needs to determine that RTB did NOT automatically Open as expected and that it also provides input to the P-4 Interlocks.

Original Question # - NRC Exam 2004-103-4
Original Question KA - 000007 EA2.03

39. 039 002/BOTH/OK/DIRECT/NRC EXAM 2006-41/012000 K6.02/2.9/3.1/H/3

Given the following conditions on Unit 1:

- A reactor startup is in progress.
- The reactor is critical in the source range.
- N41 Power Range channel has failed and been removed from service with all bistables placed in the trip condition.
- A loss of power to the CRID 2 bus occurs.

Which ONE of the following actions will occur?

A. Reactor trips and N32 Source Range channel is de-energized.
N31 Source Range channel is still in operation.

B. The reactor is critical and BOTH source range channels are de-energized.

C. The reactor is critical and N32 Source Range channel is de-energized.
N31 Source Range channel is still in operation.

D.✓ Reactor trips and BOTH source range channels are de-energized.

ANSWER: D

A - INCORRECT. P-10 will be met, both SR's will de energize.
B - INCORRECT. Reactor trips on a number of PR/SR trip setpoints.
C - INCORRECT. Reactor trips on a number of PR/SR trip setpoints. Also, P-10 will turn off both SR's.
D - CORRECT. A loss of CRID 2 causes a loss of power to N42. This loss also causes a loss of power to RPS channel 2. This will cause a trip condition for Power range trips for channel 2. Since N41 is already removed from service its bistable are in the tripped condition. This meets the 2/4 logic to cause a reactor trip. Additionally the signal for 2/4 power range channels above P-10 will cause the SR channels to deenergize.

LESSON PLAN/OBJ: RO-C-01101/#6, RO-C-01300/#RO-C-01300-E12
REFERENCE: RO-C-01100, TP-52; SOD-01300-002; SOD-01300-004

KA - 012000 K6.02
Reactor Protection System
Knowledge of the effect of a loss or malfunction of the following will have on the RPS:
Redundant channels
RO - 2.9 SRO - 3.1
CFR - 41.7 / 45.7

KA Justification - Question tests the knowledge of how the loss of redundant Channels (N42 and N32) will impact the RPS (RX Trip and De-energize the other SR channel N31).

Original Question # - RO26 Audit-16, Cook 2006 NRC Exam -41 Question AUDIT RO22-BOTH-23 Q#20
Original Question KA - SYS 015K4.01

40. 040 002/BOTH/OK/DIRECT/MASTER 01056C0007-8/013000 K4.04/4.3/4.5/F/3

Given the following conditions on Unit 2:

- Reactor power is 3%
- East Main Feedwater Pump is in service
- Both MDAFW Pumps have been stopped with control switches in NEUTRAL

Which ONE of the following signals will cause an automatic start of the MDAFW Pumps?

A. AMSAC
East Main Feedwater Pump Trip

B. Safety Injection
East Main Feedwater Pump Trip

C.✓ Safety Injection
Blackout Sequence

D. Blackout Sequence
Steam Generator Low Level of 26% on 1 of 4 SGs

ANSWER: C

A - INCORRECT. AMSAC Bypassed at <40% power. Plausible as both are auto start features of AFW but they are not in service with the described plant conditions.
B - INCORRECT. MFP Auto Start only available in AUTO. Plausible as both are auto start features of AFW but they are not in service with the described plant conditions.
C - CORRECT. Safety Injection and Blackout will start AFW Pps in Neutral or AUTO.
D - INCORRECT. Requires 1/4 SG Levels low-low (<22%) for AUTO start. Plausible as both are auto start features of AFW but one of the signals set points is not correct.

LESSON PLAN/OBJ: RO-C-05600/#7
REFERENCE: RQ-C-KNOW

KA - 013000 K4.04
Engineered Safety Features Actuation System (ESFAS)
Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following:
Auxiliary feed actuation signal
RO - 4.3 SRO - 4.5
CFR - 41.7

KA Justification - Question tests knowledge of which ESFAS signals (Interlocks) will cause AFW actuation.

Original Question # - Master Bank 01056C0007-8
Original Question KA - Unknown

41. 041 002/BOTH/OK/DIRECT/NRC EXAM 2006-002-5/014000 A4.01/3.3/3.1/H/3

During a power ascension, with reactor power at 48%, Control Bank C - Group 1 rod B-8 drops. Prior to the drop it was at 230 steps. While restoring the rod, a control rod urgent failure alarm occurs.

Which one of the following explains why the alarm actuated?

A. All other Bank C - Group 1 rod lift coil disconnect switches are open.

B.✔ All Bank C - Group 2 rod lift coil disconnect switches are open.

C. The step counter of the pulse to analog (P/A) converter was not reset to 0.

D. Group C rod moving with group D rods withdrawn.

ANSWER: B

A - INCORRECT. While all other Bank C Group 1 rods lift coils deenergized, the Alarm is generated from the failure of Group 2 movement (System monitors current through the lift coils - Since Bank C group 1 rod B-8 still has current the alarm is from group 2)

B - CORRECT. Since the dropped rod is completely inserted, the lift coil disconnect switches for all operable rods within the affected bank are opened. An Urgent failure will occur when the misaligned rod begins to move. This is caused by the non-movement of the group without the misaligned rod.

C - INCORRECT. While the P/A Converter is reset during rod recovery, failure to do so would not cause an urgent failure.

D - INCORRECT. Group C is moved in the bank select mode. This would not cause an urgent failure alarm.

LESSON PLAN/OBJ: RO-C-AOP-D8/#Ro-C-AOP0240412-E1, RO-C-01200 /#4
REFERENCE: 2-OHP-4024-210, Annunciator #210 Response: Flux Rod, Drop 26 Rod Control Urgent Failure, 2-OHP-4022-012-005

KA - 014000 A4.01
Rod Position Indication System (RPIS)
Ability to manually operate and/or monitor in the control room:
Rod selection control
RO - 3.3 SRO - 3.1
CFR - 41.7 / 45.5 to 45.8

KA Justification - Requires the ability to monitor and verify proper rod position response during a dropped rod recovery based on the rod bank/disconnect switch alignment (rod selection control).

Original Question # - Cook 2006 NRC Exam -002-5, INPO # 27278 Ginna 1-4/27/2004
Original Question KA - 000003 AA1.02, 000003AK2.05

42. 042 004/BOTH/OK/DIRECT/MASTER AOP1CAOP2.1-4/016000 K1.04/2.7/2.7/F/3

Which set of the following describes the response of the reactor protection system to a controlling feedwater flow instrument failing low with no operator action from 100% power conditions?

1) Turbine/Reactor trip on Low-Low level in associated steam generator
2) Turbine/Reactor trip on High-High level in associated steam generator
3) Feedwater Conservation
4) Feedwater Isolation

A. 2 **AND** 3

B. 1 **AND** 3

C.✓ 2 **AND** 4

D. 1 **AND** 4

ANSWER: C

A - INCORRECT. The Turbine will trip on High-High level but a FW conservation signal will not be received. Plausible since the first portion of the distractor is correct and if student believes failure initiates feedwater conservation (logical since indication shows low feedflow) then this is a logical answer.

B - INCORRECT. The SG level will not go low. Plausible since the FW flow goes low and a steam flow failure would cause this response. Plausible if student believes failure actually results in feedwater flow reduction (logical since indication shows low feedflow) then this is a logical answer.

C - CORRECT. The FW valves will open when the FW flow instrument fails low, causing actual level to rise to the High-High SG setpoint causing a Turbine trip and FW Isolation.

D - INCORRECT. The SG level will not go low. Plausible since the FW flow goes low and a steam flow failure would cause this response. Plausible if student believes actual reduction in feedwater flow for first portion of distractor and second portion of distractor occurs in a Steam Flow transmitter failure (which is very similar failure).

LESSON PLAN/OBJ: RO-C-AOP-D10/#RO-C-AOP0390412-E1
REFERENCE: SOD-05100-001, SGWLC

KA - 016000 K1.04
Non-Nuclear Instrumentation System (NNIS)
Knowledge of the physical connections and/or cause-effect relationships between the NNIS and the following systems:
MFW System
RO - 2.7 SRO - 2.7
CFR - 41.2 to 41.9 / 45.7 to 45.8

KA Justification - Question tests knowledge of NNIS (SG level & Flow) response & actuations caused by a FW system malfunction.

Original Question # - Master Bank MASTER AOP1CAOP2.1-4, CM-1127-31944

43. 043 003/BOTH/OK/DIRECT/NRC EXAM-2006-045-5/022000 K3.02/3.0/3.3/H/3

Given the following conditions:

- Unit 2 is in Mode 3 at rated temperature and pressure awaiting a startup.
- Lower Containment Cooling NESW supply is throttled to all ventilation units.
- A power failure causes a loss of 4 of the 8 Lower Containment Vent units.
- Average containment temperature rises from 100°F to 119°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 the rise in Containment temperature?

Density lowering 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: A

A - 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 rises the density & weight of the reference leg lowers. This means that the level in the pressurizer will indicate higher for the same initial actual level.

B - INCORRECT. Indicated level will be higher than actual level.

C - INCORRECT. Reference leg density lowers.

D - INCORRECT. Indicated level will be higher than actual level. Reference Leg density lowers.

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

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

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

44. 044 001/BOTH/OK/NEW/NEW/025000 A1.03/2.5/2.5/H/3

Given the following conditions on Unit 1:

- Unit is in Mode 1 at 100% power
- A control air leak has resulted in isolation of glycol to containment.
- WIN team states that it will take approximately 6 hours to repair the control air leak.

Which ONE of the following describes the operating implications of the loss of glycol to containment.

A. Immediately declare the Ice Bed Inoperable. Monitor Ice Bed temperatures to ensure they remain ≤ 27°F every 4 hours.

B.✔ Monitor Ice Bed temperatures to ensure they remain at an acceptable range. Enter appropriate Tech Spec actions if Ice Bed temperature rises to > 27°F

C. Start all Unit 1 Air Handling Units (AHUs). Monitor Ice Bed Temperature locally once per hour until glycol system is restored.

D. Maximize Containment cooling. If glycol cannot be restored within ONE hour declare the Ice Bed Inoperable.

ANSWER: B

A. INCORRECT. Technical Specifications requires that temperatures are maintained ≤27°F but the loss of glycol alone does not require Tech Spec entry

B. CORRECT. Monitoring temperatures to ensure that they remain ≤27 is all that is required. The loss of glycol alone does not require Tech Spec entry

C. INCORRECT. The AHUs are generally stopped if glycol is lost, They would NOT be started. Monitoring temperatures to ensure that they remain ≤27 is required but a one hour frequency is NOT required and temperatures may be monitored from the Control Room.

D. INCORRECT. Maximizing Containment Cooling may help slightly but is not required. Technical Specifications requires that temperatures are maintained < 27°F but the loss of glycol alone does not require Tech Spec entry

Note: **The Ice Condenser is sufficiently subcooled and insulated such that a significant temperature rise will not be observed for several days following the loss of cooling.**

LESSON PLAN/OBJ: RO-C-TS01/#11
REFERENCE: TS 3.6.11; SD-01000, pages 9 &10

KA - 025000 A1.03
Ice Condenser System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Ice Condenser System controls including:
Glycol flow to ice condenser air handling units
RO - 2.5 SRO - 2.5
CFR - 41.5 / 45.5

KA Justification - Questions tests operator knowledge of what actions are required and what monitoring is required based on loss of glycol cooling. There is also an element of prediction in that the operator needs to predict the rate of temperature rise and realize that immediate TS actions are not required.

Original Question # - NEW
Original Question KA - NEW

45. 045 001/BOTH/OK/NEW/NEW/026000 K2.02/2.7/2.9/H/3

Given the following conditions on Unit 1:

- A Large Break LOCA has occurred 10 minutes ago
- 600 VAC buses 11C and 11D de-energized on the trip
- All other systems function as desired.
- Containment pressure is 5.0 psig and rising

Which ONE of the following describes the current status of the CTS Pump discharge valves?

A. ALL CTS Pump Discharge valves are OPEN

B.✓ IMO-210, East CTS Pump Discharge is CLOSED
IMO-211, East CTS Pump Discharge is CLOSED
Both West CTS Pump Discharge valves are OPEN

C. IMO-220, West CTS Pump Discharge is CLOSED
IMO-221, West CTS Pump Discharge is CLOSED
Both East CTS Pump Discharge valves are OPEN

D. IMO-211, East CTS Pump Discharge is CLOSED
IMO-210, East CTS Pump Discharge is OPEN
IMO-221, West CTS Pump Discharge is CLOSED
IMO-220, West CTS Pump Discharge is OPEN

ANSWER: B

A - INCORRECT. The Loss of 600VAC Bus 11D will cause the discharge valves for the East CTS pump to lose power will prevent them from opening. This would be true if the valves were initially open.

B - CORRECT. The Loss of 600VAC Bus 11D will cause the discharge valves for the East CTS pump to lose power will prevent them from opening.

C - INCORRECT. The East Valves will not be open. This would be true if Bus 11A was lost

D - INCORRECT. Only the East train valves have lost power. This is plausible since the CTS pumps have two discharge valves in parallel and the operators may assume they have crossed power supplies to ensure a flowpath.

LESSON PLAN/OBJ: RO-C-00900/RO-C-00900-E9, RO-C-00900-E12
REFERENCE: RO-C-00900 pg. 10 & 20, SOD-00900-001

KA - 026000 K2.02
Containment Spray System (CSS)
Knowledge of bus power supplies to the following:
MOVs
RO - 2.7 SRO - 2.9
CFR - 41.7

KA Justification - Question tests knowledge of which CTS MOVs are impacted by the loss of a bus power supply.

Original Question # - NEW
Original Question KA - NEW

46. 046 002/BOTH/OK/NEW/NEW/029000 A2.03/2.7/3.1/H/3

Given the following conditions in Unit 1:

- Unit is in Mode 5
- Containment pressure is negative 0.3 psig.
- RP has requested that Containment Purge be placed in service.

1-OHP-4021-028-005, Operation Of The Containment Purge System, provides a defined sequence of operation due to a concern with Containment Pressure.

Which ONE of the following describes this sequence and the reason for the concern while starting up the Containment Purge System?

A. Start one Purge Supply fan then open the supply fan valve since Technical Specifications require Containment pressure to be < 0 PSIG at all times.

B.✔ Prior to starting the fans, open the Upper Containment Purge Supply valves to prevent Ice Condenser doors from opening when initiating containment purge.

C. Prior to starting the fans, open the Lower Containment exhaust fan valves to prevent Ice Condenser doors from buckling when initiating containment purge.

D. Start one Purge Exhaust fan then open the exhaust fan valve to prevent a positive pressure from adversely affecting the radiation monitor operations.

ANSWER: B

A - INCORRECT. T.S. 3.6.1.4, Internal Pressure requires pressure to be -1.5 psig to .03 psig. This action will not maintain pressure low.

B - CORRECT. A low pressure in upper containment with respect to lower containment will cause the Ice Condenser Doors to open. 1-OHP-4021-028-005 Attachment 1 step 4.7.4 is performed to raise/equalize upper containment pressure.

C - INCORRECT. The buckling concern for the Ice Condenser doors is due to uneven floor cooling not ventilation fan operation.

D - INCORRECT. The radiation monitors will not be affected by minor pressure variations.

LESSON PLAN/OBJ: RO-C-02800/RO-C-02800-T1
REFERENCE: 1-OHP-4021-028-005, Operation Of The Containment Purge System, Attachment 1

KA - 029000 A2.03
Containment Purge System (CPS)
Ability to (a) predict the impacts of the following malfunctions or operations on the Containment Purge System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Startup operations and the associated required valve lineups
RO - 2.7 SRO - 3.1
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - The question tests the ability of the operator to determine to correct procedural actions required and the consequences of not following those actions (Predicts impacts of incorrect purge operations and prevents impacts through correct sequence)

Original Question # - NEW
Original Question KA - NEW

47. 047 002/BOTH/OK/NEW/NEW/035000 K6.02/3.1/3.5/H/3

Given the following condition on Unit 2:

- Unit is operating at 100% power when an inadvertent Steam Line Isolation occurred.
- Immediately following the isolation, and resultant plant response, the operators note that all the SG PORVs failed to open.

One minute following the Steam Line Isolation, which ONE of the following describes the maximum expected SG pressures?

A. 1025 psig

B. 1040 psig

C.✓ 1065 psig

D. 1085 psig

ANSWER: C

A - INCORRECT. This is the Normal PORV Setpoint and would be the expected pressures of the other SGs.
B - INCORRECT. This is the PORV setpoint used in the SG tube rupture procedures for the faulted SG.
C - CORRECT. Following the Steam Line Isolation a Rx trip would be expected due to OTDT. SG pressures would initially surge opening most of the safeties but as the RCS cooled down pressures would stabilize on the lowest safety valve setpoint (1065 psig) due to the reduction in Reactor Power and Decay heat during the initial 30 seconds of the event.
D - INCORRECT. The pressures may initially surge to this level but would quickly (less than 30 seconds) drop after the Rx trip.

Note - SG Safety valve setpoints:

SV-1A	**1065 psig**
SV-1B	**1065 psig**
SV-2A	**1075 psig**
SV-2B	**1075 psig**
SV-3	**1085 psig**

LESSON PLAN/OBJ: RO-C-05103/#RO-C-05103 -E2, #RO-C-05103-E6
REFERENCE: RO-C-05103 pg. 11-12

KA - 035000 K6.02
Steam Generator System (S/GS)
Knowledge of the effect of a loss or malfunction of the following will have on the SGs:
Secondary PORV
RO - 3.1 SRO - 3.5
CFR - 41.7 / 45.7

KA Justification - Requires knowledge of how the loss of a PORV and subsequent SG Stop Valve closure will impact the SG pressure.

Higher order based on a requirement to determine that the Rx will trip on SLI due to OTDT.

Original Question # - NEW
Original Question KA - NEW

48. 048 004/BOTH/OK/DIRECT/MAST 01RXTC1021-13/039000 K5.08/3.6/3.6/F/2

Given the following conditions:

- Reactor power is 50% with a negative moderator temperature coefficient
- Control Rods are in manual.

If steam flow is raised by 5 percent, which ONE of the following statements best describes how reactor power will respond to the change?

Reactor power will:

A. decrease to a new lower value.

B. increase temporarily, then return to its initial value.

C✓ increase to a new higher value.

D. decrease temporarily, then return to its initial value.

ANSWER: C

A - INCORRECT. This would be the response of RCS temperature
B - INCORRECT. This would be true for a rise in Rod position
C - CORRECT. The increased Steam flow will cause RCS temperature to lower and add positive reactivity due to the negative MTC
D - INCORRECT. This would be true for a rise Boron Concentration

LESSON PLAN/OBJ: RO-C-GF10/#21
REFERENCE: RO-C-GF10

KA - 039000 K5.08
Main and Reheat Steam System (MRSS)
Knowledge of the operational implications of the following concepts as they apply to the MRSS:
Effect of steam removal on reactivity
RO - 3.6 SRO - 3.6
CFR - 41.5 / 45.7

KA Justification - Question tests knowledge of power rise (operational Implication) due to a rise in steam flow causing positive reactivity feedback.

Original Question # - Master Bank 01RXTC1021-13

49. 049 002/BOTH/OK/DIRECT-REPEAT/NRC EXAM 2007-52/041000 K2.01/2.8/2.9/F/3

Which ONE of the following power supply failures would allow the steam dump system to continue to operate?

A. CRID II

B.✓ CRID III

C. 250 VDC Bus VDAB

D. 250 VDC Bus VDCD

ANSWER: B

A - INCORRECT. CRID II powers the Steam Dump Controllers
B - CORRECT. CRID III does Not supply power to the Steam Dumps or relays.
C - INCORRECT. 250 VDC Bus VDAB powers 1 train of Steam Dump Solenoids
D - INCORRECT. 250 VDC Bus VDCD powers 1 train of Steam Dump Solenoids

LESSON PLAN/OBJ: RO-C-05200/#4
REFERENCE: RO-C-05200 Steam Dump System pg. 15-16

KA - 041000 K2.01
Steam Dump System (SDS) and Turbine Bypass Control
Knowledge of bus power supplies to the following:
ICS, normal and alternate power supply
RO - 2.8 SRO - 2.9
CFR - 41.7

KA Justification - Question tests knowledge of all of the Steam dump power supplies by requiring the operator to identify the one that doesn't supply power.

Original Question # - Master Bank 01052C0002-4, NRC EXAM 2007-52
Original Question KA - 041000 K2.01

50. 050 004/BOTH/OK/MODIFIED/PRARILND-4232004-14/056000 2.2.44/4.2/4.4/F/3

Unit 1 has just completed a Heatup and is preparing for a Reactor Startup.

You have been directed to open the FW pump Emergency Leak Offs (ELOs).

Which ONE of the following describes the indications/systems that you should check prior to opening these valves and why?

A. Ensure that the FW pump is reset to allow the ELOs to be opened.

B. Ensure that the FW pump oil system is operating and has been warmed to minimize the effects of cold seal water on FW pump bearings.

C.✓ Ensure that the FW pump oil system is operating to prevent damage to the FW pumps due to condensate flow spinning the pumps.

D. Ensure that the FW pump has been removed from turning gear to prevent damage to the turning gear motor.

ANSWER: C

A - INCORRECT. The FW turbine needs to be tripped (Stop valve closed) to allow the ELOs to be fully closed. They will position based on flow if the stop valve is opened. Plausible as this is a true statement but not required for the action being described.

B - INCORRECT. The oil system operation would have minimal impact on the seal water temperature. Plausible as the oil system is required to be in service for this evolution but not for the described purpose.

C - CORRECT. Placing flow though the FW pumps (opening recirculation valves) causes the turbine and pump to rotate at > 100 rpm and so the oil system is required for bearing protection.

D - INCORRECT. The FW pump will roll off the turning gear (become disengaged) when the ELOs are opened and the motor will not be damaged. Plausible to prevent equipment damage.

LESSON PLAN/OBJ: RO-C-05500-E8, RO-C-NOP7-E2
REFERENCE: 2-OHP-4021-055-003

KA - 056000 2.2.44
Condensate System
Equipment Control
Ability to interpret control room indications to verify the status and operation of a 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

KA Justification - Question tests ability to determine what the condition of the systems must be prior to aligning the FW ELOs and how the required lineup impacts the equipment.

Original Question # - PRARILND-4232004-14
Original Question KA - 056 K1.03

51. 051 004/BOTH/OK/DIRECT/RO26 AUDIT-41/059000 A1.07/2.5/2.6/H/3

The Unit 2 FW Pump Discharge Header Pressure Transmitter 2-FPC-250A slowly drifts LOW during normal plant operation.

This will cause the MFP Speed Control System to generate an indicated FW Delta-P signal ____(1)____ than required, causing the main feed pump(s) to ____(2)____.

Note: Assume FPC-250A is not identified as failed by DCS.

	(1)	(2)
A.	larger	speed up
B.	larger	slow down
C.✓	smaller	speed up
D.	smaller	slow down

ANSWER: C

A - INCORRECT. Steam to FW discharge pressure DP will be smaller. Plausible if the student does not know how the DP is being derived.

B - INCORRECT. Steam to FW discharge pressure DP will be smaller. The controller will raise FW pump Speed. Plausible if the student does not know how the DP is being derived.

C - CORRECT. The Main FW Pump Speed control compares the UPC-102A/B (highest) steam header pressure to the FW pump Discharge pressure FPC-250A/B (lowest). The speed control attempts to maintain the Main FW Pump speed such that the FW header to Steam Header DP is on Program. When the FW Discharge Pressure drifts Low, it will appear that a smaller DP exists which will raise FW pump speed to try to raise FW pump Discharge header pressure.

D - INCORRECT. The controller will raise FW pump Speed. Plausible if the student does not know how the impact of DP effects SGFP controls.

LESSON PLAN/OBJ: RO-C-05501/#RO-C-05501-E5
REFERENCE: TS3000, DCS BODD, Page 4-7

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 - 3SPK

K/A Justification - Question tests ability to predict changes associated with the FW Pump Speed.

Original Question # - Modified from NRC EXAM 2007-054 (UPC failure to FPC failure, Failure to Low & Updated due to DCS), Cook 2006 NRC Exam -COOK06-54 , Master Bank 01055C0008-5, RO26 AUDIT-41
Original Question KA - 059 K4.05

52. 052 003/BOTH/OK/DIRECT/RO24 AUDIT 044-12/059000 A4.03/2.9/2.9/F/2

Which ONE of the following describes the functional relationship with respect to controlling steam generator (SG) levels between the Main Feedwater Pumps (MFPs) and the Main Feedwater Regulating Valves (MFRVs) when the unit is ramping from 50% to 100% power?

A.✓ The MFPs maintain a variable differential pressure across the MFRVs, while the MFRVs throttle to maintain a constant SG water level.

B. The MFPs maintain a constant differential pressure across the MFRVs, while the MFRVs throttle to maintain a variable SG water level.

C. The MFPs maintain a variable differential pressure across the MFRVs, while the MFRVs throttle to maintain a variable SG water level.

D. The MFPs maintain a constant differential pressure across the MFRVs, while the MFRVs throttle to maintain a constant SG water level.

ANSWER: A

A - CORRECT. The design of the SGWLC system is to maintain a constant level in the SGs at all power levels. The MFW control system however varies the programming to maintain an optimum DP across the MFRVs.

B - INCORRECT. The DP is not constant it varies with program while the level is held constant. Plausible due to second portion of the distractor being correct and the student must know the SGFP varies the D/P

C - INCORRECT. The Level is held constant. Plausible as the first portion of the distractor is correct and the student must know SG level is constant for all power levels which is unique in Westinghouse plants

D - INCORRECT. The DP is not constant it varies with program. Plausible as the second portion of the question is correct and the student must know the SGFP varies the D/P.

LESSON PLAN/OBJ: RO-C-05100/#3&6
REFERENCE: SOD-05100-001, RO-C-05100 Steam Generator System

KA - 059000 A4.03
Main Feedwater (MFW) System
Ability to manually operate and/or monitor in the control room:
Feedwater control during power increase and decrease
RO - 2.9 SRO - 2.9
CFR - 41.7 / 45.5 to 45.8

KA Justification - Requires knowledge of how to monitor and control feedwater flow to the SGs during a power escalation.

Original Question # - RO24 AUDIT 044-12
Original Question KA - 059 A3.02

53. 053 002/BOTH/OK/DIRECT/RO26 AUDIT-44/061000 K3.02/4.2/4.4/H/3

Given the following conditions on Unit 2:

- Unit is in Mode 1.
- The TDAFW Pump is tagged out of service.
- A Loss of Feedwater causes a reactor trip.
- Coincident with the trip, T21D Differential trip actuates.

Which ONE of the following describes the Auxiliary Feedwater alignment and approximate flow rates?

A.✓ 1 and 4 SGs being fed at 120 x 10^3 pph each

B. 1 and 4 SGs being fed at 240 x 10^3 pph each

C. 2 and 3 SGs being fed at 120 x 10^3 pph each

D. ALL SGs being fed at 120 x 10^3 pph each

ANSWER: A

A - CORRECT. The T21D Differential causes a loss of T21D Bus. With a loss of T21D, Only the West MDAFW Pump is available. Capacity is ~240 x 10^3 pph, and it is aligned to automatically feed 1 and 4 SGs.

B - INCORRECT. Capacity of TDAFW aligned to 2 SGs

C - INCORRECT. West MDAFW would be aligned to 1 & 4 SGs, not 2 & 3 SGs (East)

D - INCORRECT. This would be the alignment if the TDAFW Pump was the only operating pump.

LESSON PLAN/OBJ: RO-C-05600/#2 & 3
REFERENCE: RO-C-05600 Auxiliary Feedwater System pg. 12-14, SOD-08201-001

KA - 061000 K3.02
Auxiliary / Emergency Feedwater (AFW) System
Knowledge of the effect that a loss or malfunction of the AFW System will have on the following:
S/G
RO - 4.2 SRO - 4.4
CFR - 41.7 / 45.6

K/A Justification - Question asks candidate to identify the amount of AFW flow to the specific Steam Generators from a single AFW pump.

Original Question # - RO26 AUDIT-44 from SEQ2007
Original Question KA - 061000 K6.02

54. 054 006/BOTH/OK/DIRECT/NRC EXAM 2006-057-5/062000 K1.03/3.5/4.0/H/3

The operator incorrectly opens the breaker labeled "7.5 KVA Static Inverter Channel IV" on 250 VDC distribution panel "MCAB". The operator realizes the mistake and immediately recloses the breaker.

Which ONE of the following describes the effect of these actions, if any?

A. The alternate power source to the CRID Inverter will be lost when the breaker is reclosed. The CRID will transfer to the 120 VAC from the Regulating Transformer.

B. The alternate power source to the CRID Inverter will be lost. No automatic action will occur when the breaker is reclosed. The auto transfer lockout must be reset at the inverter.

C.✓ The normal power source to the CRID Inverter will be lost so it will auto transfer to the alternate source. When the breaker is reclosed, it will auto transfer to the normal source.

D. The normal power source to the CRID Inverter will be lost so it will auto transfer to the alternate source. When the breaker is reclosed, the auto transfer lockout must be reset at the inverter.

ANSWER: C

A - INCORRECT. The Alternate source will not be lost. The normal DC supply will be restored and the Inverter will re-transfer to the normal source. Plausible due to continuity of power remains to the panel through out the evolution

B - INCORRECT. The Alternate source will not be lost. The normal DC supply will be restored and the Inverter will re-transfer to the normal source. Plausible assuming student believes the panel will lose power upon inadvertent operation of the supply breaker which is a valid assumption with a DC vital breaker.

C - CORRECT. The static transfer switch provides a virtual zero time transfer to the alternate source in case of inverter failure. Thirty seconds after the static switch transfer event ceases and all system parameters are normal, the static switch automatically re-transfers the load to the inverter, without power interruption.

D - INCORRECT. The normal DC supply will be restored when the breaker is closed and the Inverter will re-transfer to the normal source. Plausible as the restoration of power is correct if the logic contained and auto transfer lockout which does exist on several other plant electrical components.

LESSON PLAN/OBJ: RO-C-08203/#3e, #3g
REFERENCE: RO-C-08203, Instrumentation Electrical System

KA - 062000 K1.03
A.C. Electrical Distribution System
Knowledge of the physical connections and/or cause-effect relationships between the A.C. Distribution System and the following systems:
DC distribution
RO - 3.5 SRO - 4.0
CFR - 41.2 to 41.9 / 45.7 to 45.8

KA Justification - Question tests for knowledge of how the DC system connects and supports the AC distribution system (feeds CRID) and the cause-effect relationship due to breaker manipulations.

Original Question # - Cook 2006 NRC Exam - 057-5, Bank 01082C0303-2, CM-7852-38509
Original Question KA - 063 K4.01, 062000 A3.04

55. 055 005/BOTH/OK/DIRECT/DIABLO2007-050-1/063000 K4.04/2.6/2.9/H/2

Which ONE of the following describes the effect on a closed Circulating Water pump breaker if DC control power is lost to the breaker?

A. The breaker immediately trips open and cannot be reclosed until control power is restored.

B. The breaker can be tripped from the Control Room but automatic trip functions are not operable.

C.✓ Automatic trips are not operable and tripping the breaker from the Control Room is not possible.

D. Automatic breaker trips are operable but tripping the breaker from the Control Room is not possible.

ANSWER: C

A - INCORRECT. The breaker has stored energy in the spring, but it can not be released due to the loss of power. Plausible since many signals (RPS) require power to maintain contacts open.

B - INCORRECT. The breaker has stored energy in the spring, but it can not be released due to the loss of power. Plausible since many signals(RPS) require power to maintain contacts open and generate trip on loss of power.

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

D - INCORRECT. While it is true that the spring has stored energy, the spring release mechanism can not release the spring to cause the trip.

LESSON PLAN/OBJ: RO-C-05700-E11, RO-C-08204-E1, RO-C-AOP0550412-E1
REFERENCE: RO-C-05700 TP-36

KA - 063000 K4.04
D.C. Electrical Distribution System
Knowledge of D.C. Electrical System design feature(s) and/or interlock(s) which provide for the following:
Trips
RO - 2.6 SRO - 2.9
CFR - 41.7

KA Justification - Question requires knowledge of the design features provided by DC power for generating a Trip signal.

Original Question # - DIABLO2007-050-1
Original Question KA - 063 K4.04

56. 056 003/BOTH/OK/DIRECT/RO26 AUDIT-48/064000 2.4.50/4.2/4.0/H/3

Given the following conditions on Unit 1:

- Bus T11B normal supply breaker has opened.
- DG1AB Diesel Generator has started and is tied to the bus.
- Ann. 118, Drop 53, DG1AB TRIPS DISABLED is LIT

Which ONE of the following conditions will automatically trip the diesel generator?

A.✓ Engine Speed of 590 rpm

B. CO_2 actuating in the EDG Room

C. Main Bearing Temperature 198° F

D. Low Lube Oil Pressure 23 psig

ANSWER: A

A - CORRECT. EDG is in Emergency Mode so Overspeed Trip is the only one available, 590 rpm is 114.7% of Normal 514 rpm - Trip at 110%.

B - INCORRECT. CO_2 is trip but not in emergency mode.

C - INCORRECT. Main bearing temp of >195 is normal trip

D - INCORRECT. Lube oil pressure of <25 psig is normal trip

Note: LOOP or SI places EDG in Emergency Mode and blocks 7 non-emergency trips.

LESSON PLAN/OBJ: RO-C-03200/#10
REFERENCE: RO-C-03200 Emergency Diesel Generators

KA - 064000 2.4.50
Emergency Diesel Generator (ED/G) System
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

KA Justification - Requires the ability to determine proper diesel trip setpoint and understand (monitor) conditions that will trip the diesel generator for the given plant conditions.

Original Question # - RO23 Audit -059-5 (Q#54), RO26 Audit-48
Original Question KA - 064000 K4.01

57. 057 001/BOTH/OK/NEW/NEW/064000 K4.04/3.1/3.7/F/3

Which ONE of the following describes a condition that would cause a load conservation signal to be generated for the DG1AB.

A. Train A SI and Train A Load Shed

B. Train B SI and Train B CTS

C. Train A SI and Train A CTS

D.✓ Train B SI with a Loss of Offsite Power

ANSWER: D

A - INCORRECT. Train A would cause a Load Conservation on DG1CD.
B - INCORRECT. Should be LOOP or Load Shed with SI. Plausible since starting CTS provides additional loads and NESW Pump response is different with CTS actuation.
C - INCORRECT. Train A would cause a Load Conservation on DG1CD. Should be LOOP or Load Shed with SI. Plausible since starting CTS provides additional loads and NESW Pump response is different with CTS actuation.
D - CORRECT. An SI with a LOOP (and subsequent Load Shed) will generate a Load Conservation signal. Train B is associated with DG1AB.

LESSON PLAN/OBJ: RO-C-08201/#RO-C-08201-E6
REFERENCE: RO-C-08201

KA - 064000 K4.04
Emergency Diesel Generator (ED/G) System
Knowledge of ED/G System design feature(s) and/or interlock(s) which provide for the following:
Overload ratings
RO - 3.1 SRO - 3.7
CFR - 41.7

KA Justification - A load conservation signal is generated to prevent EDG overloading. This question requires knowledge of the design feature (load conservation) that prevents overloading of the EDG.

Original Question # - New
Original Question KA - New

58. 058 002/BOTH/OK/DIRECT/RO25 AUDIT-59/068000 K4.01/3.4/4.1/F/3

Which ONE of the following lists the two conditions that will independently cause termination of a liquid release by closing 12-RRV-285, Liquid Waste Disposal Effluent Discharge Header Shutoff Valve, and/or tripping the operating monitor tank pump?

A. Low circulating water flow
High radiation sensed in the release header

B. Low circulating water flow
High radiation sensed in the circulating water flow

C.✓ Low release header radiation monitor sample flow
High radiation sensed in the release header

D. High release header radiation monitor sample flow
High radiation sensed in the circulating water flow

ANSWER: C

A - INCORRECT. Even though there is a requirement to have adequate circulating water flow, there is no trip for RRV-285 due to low flow conditions.
B - INCORRECT. Even though there is a requirement to have adequate circulating water flow, there is no trip for RRV-285 due to low flow conditions. The radiation monitor senses radiation levels on the actual release line, not the Circ Water system.
C - CORRECT. RRS-1001 High alarm sensed on the actual release line will energize R18-AUX & R18-AUX1 which closes RRV-285 and trips the monitor tank pumps. In addition, either high or low sample flow (less than 20% or greater than 90%) will energize R18-AUX & R18-AUX1.
D - INCORRECT. The radiation monitor senses radiation levels on the actual release line, not the Circ Water system.

LESSON PLAN/OBJ: RO-C-02200/ #5, #8
REFERENCE: 12-OHP-4024-139, Drop 18, OP-12-98810, OP-12-98313, OP-12-98276

KA - 068000 K4.01
Liquid Radwaste System (LRS)
Knowledge of Liquid Radwaste System design feature(s) and/or interlock(s) which provide for the following:
Safety and environmental precautions for handling hot, acidic, and radioactive liquids
RO - 3.4 SRO - 4.1
CFR - 41.7

KA Justification - Question tests knowledge of Liquid Radwaste System design features and interlocks which provide for the isolation of radioactive liquids to prevent excessive radioactive discharge to the environment.

Original Question # - AUDIT RO22-SRO-9, RO25 AUDIT-59
Original Question KA - 068000 K4.01 3.4/4.1 CFR 41.7

59. 059 003/BOTH/OK/MODIFIED/NRC EXAM 2007-19/072000 A3.01/2.9/3.1/F/3

Which ONE of the following lists the Unit 1 Control Room Ventilation system damper alignment for operation during a high alarm on ERS-7401, U1 Control Room Area Radiation Monitor?

	1-HV-ACR-DA-1/1A Outside air to CR	**1-HV-ACR-DA-2** Outside air to CR PRZN	**1-HV-ACR-DA-2A** Outside air to CR PRZN	**1-HV-ACR-DA-3** CR air to PRZN
A.	OPEN	CLOSED	PARTIAL OPEN	OPEN
B.✓	CLOSED	PARTIAL OPEN	CLOSED	OPEN
C.	OPEN	PARTIAL OPEN	CLOSED	CLOSED
D.	CLOSED	CLOSED	PARTIAL OPEN	CLOSED

ANSWER: B

A - INCORRECT. Dampers 1/1A will be closed on an ERS-7401 high alarm.
B - CORRECT. On an ERS-7401 high alarm: Damper 1/1A will be closed; Damper 2 will be partially open; Damper 3 opens.
C - INCORRECT. Damper 1/1A will be closed and Damper 3 will remain open on an ERS-7401 high alarm.
D - INCORRECT. Damper 3 will remain open on an ERS-7401 high alarm.

LESSON PLAN/OBJ: RO-C-02801A/#8
REFERENCE: SOD-02801A-001

Modified: Changed stem to radiation alarm (vs. fire) which changed the correct answer to B (vs. D)

KA - 072000 A3.01
Area Radiation Monitoring (ARM) System
Ability to monitor automatic operation of the ARM system, including:
Changes in ventilation alignment
RO - 2.9 SRO - 3.1
CFR - 41.7 / 45.5

KA Justification - Question tests ability to monitor changes in the Control Room Ventilation dampers caused by a high radiation alarm.

Original Question # - MASTER 01028C01A02-6, NRC EXAM 2007-19
Original Question KA - 000067 AA2.02

60. 060 002/BOTH/OK/DIRECT/RO23-AUDIT-074-3/073000 A2.01/2.5/2.9/F/3

Given the following 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 power supply

Which ONE of the following describes the response of the CCW system for the given conditions and the subsequent operator actions required?

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

Notify RP of the failed CRS-4301, East CCW HX Radiation Monitor.

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

Split the CCW Trains with Misc Header on the West Train and isolate the East Train.

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

Remove CRS-4301, East CCW HX Radiation Monitor, from service and re-align CCW flow through the West CCW Hx ONLY.

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

Notify RP to remove CRS-4301, East CCW HX Radiation Monitor, from service, then reopen 2-CRV-412.

ANSWER: D

A - INCORRECT. Either radiation monitor will close the CCW vent valve, regardless of the operating pump. Plausible based on assumption that the active portion of the system is being monitored.

B - INCORRECT. An EXTERNAL FAIL in either CRS-4301 (East CCW Header) **OR** CRS-4401 (West CCW Header) will close the CCW Vent valve. Plausible based on assumption that the active portion of the system is being monitored.

C - INCORRECT. CCW Rad Monitors do not cause auto re-alignment of the CCW system. Plausible based on maintaining flow in the active portion of the system.

D - CORRECT. An EXTERNAL FAIL in either CRS-4301 (East CCW Header) **OR** CRS-4401 (West CCW Header) will close the CCW Vent valve.

LESSON PLAN/OBJ: RO-C-01600/RO-C-01600-E6
REFERENCE: 12-OHP-4024-139 #29

KA - 073000 A2.01
Process Radiation Monitoring (PRM) System
Ability to (a) predict the impacts of the following malfunctions or operations on the PRM 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 - 2.5 SRO - 2.9
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Question tests ability to predict the impact of the loss of power on a process rad monitor, and the actions the operator should take in response to the failure.

Original Question # - RO23-AUDIT-074-3
Original Question KA - SYS 073 A2.02

61. 061 002/BOTH/OK/DIRECT/01EOPC1412-1/076000 2.4.6/3.7/4.7/F/2

The operators are attempting to energize emergency bus T21A from DG2AB during a loss of all AC power.

Which ONE of the following states the equipment switch that would NOT be placed in PULL TO LOCKOUT and the reason why?

A.✓ ESW pump to ensure diesel cooling.

B. CCW pump to ensure cooling to vital loads.

C. MDAFW pump to ensure an adequate heat sink is maintained.

D. CTS pump to ensure containment integrity is not challenged.

ANSWER: A

A - CORRECT. Since the EDG is the probably source of power, ESW to the EDG needs to be available immediately to maintain the EDG cooled once it starts and is loaded.

B - INCORRECT. CCW Pumps are locked out. Plausible as students know CCW is required to maintain RCP seals cooled and prevent SBLOCA.

C - INCORRECT. MDAFW Pumps are locked out. Plausible as students know Heat sink is a significant contributor to risk.

D - INCORRECT. CTS Pumps are locked out. Plausible as students know a LOCA is a risk due to a loss of LOOP, and CTS would be required to address Containment Pressure.

LESSON PLAN/OBJ: RO-C-EOP14/12
REFERENCE: OHP-4023-ECA-0.0, ATT A

KA - 076000 2.4.6
Service Water System (SWS)
Emergency Procedures/Plan
Knowledge of EOP mitigation strategies.
RO - 3.7 SRO - 4.7
CFR - 41.10 / 43.5 / 45.13

KA Justification - Question requires knowledge of how the ESW system (SWS) is operated during power restoration during an EOP event (mitigation of a Loss of ALL AC).

Original Question # - 01EOPC1412-1
Original Question KA - EPE:055 EK3.02

62. 062 003/BOTH/OK/DIRECT/WATTSBAR-2009-52/076000 A1.02/2.6/2.6/H/3

Given the following plant conditions:

- Unit 2 is in MODE 4 at 280°F.
- The Unit 2 East CCW HX is in service with 5000 GPM of ESW Flow.
- The Unit 2 EAST RHR train in service with 5000 GPM of CCW Flow.
- All four ESW pumps are in service with crossties open.
- The DG2CD is running at a constant 3200 KW for a Surveillance test.

Which ONE of the following describes the impact on the listed parameters if the Unit 2 East ESW pump trips?

Note: Assume no operator action.

	U2 East CCW Heat Exchanger <u>CCW Outlet Temperature</u>	**DG2CD Jacket Water (JW) Heat Exchanger <u>JW Outlet Temperature</u>**
A.✓	RISES	RELATIVELY CONSTANT
B.	RISES	RISES
C.	RELATIVELY CONSTANT	RELATIVELY CONSTANT
D.	RELATIVELY CONSTANT	RISES

ANSWER: A

A - CORRECT. ESW Flow through the CCW HX is maintained by manually throttling ESW through the CCW HX. When the East ESW pump Trips this will lower flow through the HX (since the flow is manually controlled). The Flow through the DG is maintained constant by the automatic temperature control valve.

B - INCORRECT. CCW HX temperature will rise due to lower flow but DG flow and temperatures are relatively constant due to the DG temperature control Valve, plausible since the system pressure/flow will lower and student is not aware of DG temperature control valve.

C - INCORRECT. CCW HX temperature will rise due to lower flow while DG flow and temperatures are relatively constant due to the DG temperature control Valve, plausible since the system pressure/flow will lower and student may think that ESW through CCW is temperature controlled and/orstudent may assume that the West ESW would supply flow (ESW crossties are between units not trains).

D - INCORRECT. CCW HX temperature will rise due to lower flow and DG flow and temperatures are relatively constant due to the DG temperature control Valve, plausible since the system pressure/flow will lower and student may think that ESW through CCW is temperature controlled and/or student may assume that the West ESW would supply flow (ESW crossties are between units not trains).

LESSON PLAN/OBJ: RO-C-01900/RO-C-01900-E11 , RO-C-01600/ RO-C-01600-E4, RO-C-03201/RO-C-03201-E2

REFERENCE: RO-C-01900, RO-C-03201, SOD-01900-001

KA - 076000 A1.02
Service Water System (SWS)
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including:
Reactor and turbine building closed cooling water temperatures
RO - 2.6 SRO - 2.6
CFR - 41.5 / 45.5

KA Justification - Question tests ability to predict/monitor CCW & DG JW parameters/temperatures associated with reduction in ESW (SWS) cooling.

Original Question # - WATTSBAR-2009-52
Original Question KA - SYS 076 A1.02

63. 063 006/BOTH/OK/MODIFIED/NRC EXAM 2004-93-4/078000 A3.01/3.1/3.2/F/3

Procedure 2-OHP-4022-064-001, Control Air Malfunction, has been entered based upon a 100 psi Control Air Pressure alarm.

Which ONE of the following is the correct sequence of events that will occur automatically as air pressures lower?

A.✓	95 psig at PPS-10 (20)	Standby PAC starts
	90 psig CAS wet receiver pressure	CAC starts
	85 psig at PPS-11 (21)	Plant air header isolates
B.	98 psig at PPS-10 (20)	Standby PAC starts
	97 psig CAS wet receiver pressure	CAC starts
	95 psig Control Air pressure	Plant air header isolates
C.	97 psig CAS wet receiver pressure	CAC starts
	95 psig at PPS-10 (20)	Standby PAC starts
	90 psig at PPS-11 (21)	Plant air header isolates
D.	95 psig CAS wet receiver pressure	CAC starts
	90 psig at PPS-11 (21)	Plant air header isolates
	85 psig at PPS-10 (20)	Standby PAC starts

ANSWER: A

<u>AIR SYSTEM SETPOINTS</u>
125 psig Air receiver safety valves open
104 psig at PAC discharge PAC surge protection-unloader opens
100 psig CAS wet receiver pressure CAC unloads
98 psig at PPS-11(21) Plant air header un-isolates
97 psig Turbine Building air header Alarm "PAC failure / low pressure"
95 psig at XPA-100 (100# header) Control air pressure low alarm
95 psig at PPS-10 or 20 Stand-by PAC starts
93 psig CAS wet receiver pressure CAC loads
90 psig CAS wet receiver pressure Associated CAC auto-starts
85 psig at PPS-11(21) Plant air header isolates
80 psig Control Air Pressure Manual Reactor Trip

A - CORRECT. See pressures and order above
B - INCORRECT. Setpoints wrong
C - INCORRECT. Setpoints and Order wrong
D - INCORRECT. Setpoints and Order wrong

LESSON PLAN/OBJ: RO-C-06401/#4
REFERENCE: SOD-06401-002, PLANT AIR SYSTEM

KA - 078000 A3.01
Instrument Air System (IAS)
Ability to monitor automatic operation of the IAS, including:
Air pressure
RO - 3.1 SRO - 3.2
CFR - 41.7 / 45.5

KA Justification - Question tests ability to monitor automatic actions that will occur as air pressure drops.

Original Question # - Cook RO24 Audit - 054-13, NRC02-105-2, 04-93-4

64. 064 004/BOTH/OK/DIRECT/10098-DIA-1999-160/086000 A1.03/2.7/3.2/F/3

Given the following conditions on Unit 1:

- Plant heatup is in progress with RCS temperature at 420°F.
- Fire system engineer reports that the fire door at the entrance to the positive displacement charging pump and the centrifugal charging pump rooms is nonfunctional.
- Additionally the engineer reports that no other fire system impairments exist.

Which ONE of the following is the minimum required action?

A.✓ Establish an hourly fire watch patrol within 1 hour.

B. Enter Tech Spec 3.0.3 due to no operable charging pumps.

C. Verify, by inspection, the operability of the manual fire fighting equipment within 1 hour.

D. Close the door and establish a continuous fire watch on at least one side of the fire door within 15 minutes.

ANSWER: A

A - CORRECT.	Hourly fire watch is required since no other impairments exist. Equipment in the area is still operable. Action A.1.1 and A.1.2
B - INCORRECT.	Hourly fire watch is required. Equipment in the area is still operable.
C - INCORRECT.	No inspection is required. An hourly firewatch meets TRM requirements.
D - INCORRECT.	Action A.3.1 and A.3.2 could be performed (review HVAC and close door) within 1 hour OR a continuous fire watch established within 1 hour (A.4) but both are not required within 15minutes.

LESSON PLAN/OBJ: RO-C-TS01/#11
 RO-C-ADM05/#9
REFERENCE: TRM 8.7.10 Fire Rated Assemblies

KA - 086000 A1.03
Fire Protection System (FPS)
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fire Protection System controls including:
Fire doors
RO - 2.7 SRO - 3.2
CFR - 41.5 / 45.5

KA Justification - Requires the ability to predict the operability and actions required in the event that a fire door (Recip/CCP Room Door) becomes inoperable.

Original Question # - 10098-DIA-1999-160

65. 065 004/BOTH/OK/DIRECT/AOP2604-1/103000 K1.02/3.9/4.1/H/2

Given the following conditions on Unit 1:

- Reactor has tripped.
- RO notes that two large sections of the containment isolation panel valves have just gone closed.

Which ONE of the following sets of conditions would indicate that this is due to a spurious Phase A signal?

	CNTMT PRESSURE	STEAM LINE DELTA-P	SG PRESSURE	PZR PRESSURE
A.	2.9 psig	80 psid	1005 psig	2000 psig
B.	0.2 psig	20 psid	450 psig	1810 psig
C.✓	0.5 psig	15 psid	600 psig	1900 psig
D.	1.7 psig	110 psid	800 psig	1850 psig

ANSWER: C

A - INCORRECT. Containment pressure is high enough (>1.0 psig) for an SI/Phase A.
B - INCORRECT. SG Pressures are below the SI/Phase A setpoint.
C - CORRECT. All parameters listed are within values to prevent an SI/Phase A actuation.
D - INCORRECT. Containment pressure is high enough (>1.0 psig) and Steam Line Delta-P is high enough (>100 psid) for an SI/Phase A actuation.

LESSON PLAN/OBJ: RO-C-AOP-D6\#RO-C-AOP0520412-E1
REFERENCE: OHP-4022-034-003

KA - 103000 K1.02
Containment System
Knowledge of the physical connections and/or cause-effect relationships between the Containment System and the following systems:
Containment isolation/containment integrity
RO - 3.9 SRO - 4.1
CFR - 41.2 to 41.9 / 45.7 to 45.8

KA Justification - Question tests knowledge of the conditions which will cause a Containment Isolation.

Original Question # - AOP2604-1

66. 066 003/BOTH/OK/DIRECT/MAST OPMED-19/194001 2.1.4/3.3/3.8/F/3

A licensed individual is planning to undergo some medical evaluations and a test utilizing radioisotopes. It has been determined that this test will not affect judgment or fitness for duty in any way.

Which ONE of the following describes the procedural requirements for these conditions?

The licensed individual:

A. Does not need to report this condition as a potentially disqualifying condition since it is not a fitness for duty issue.

B. Must report this situation to the fitness-for-duty liaison for independent verification that it is not a fitness for duty issue, prior to assuming license duties.

C. Must notify the Plant Manager who will evaluate the condition, prior to assuming license duties.

D.✓ Must notify the Ops Training Manager of a potential disqualifying medical condition

ANSWER: D

A - INCORRECT. This condition could affect a person's ability to perform required tasks in the Aux Building..
B - INCORRECT. No independent review by the fitness -for-duty liaison is required..
C - INCORRECT. The Plant Manager is not responsible for this item. Reporting needs to be made to the Ops Training Manager.
D - CORRECT. The described condition does not affect judgment nor is it a fitness for duty issue. Since a medical test that utilizes radioisotopes would impact an individual's ability to enter and exit the auxiliary building, it limits the individual's ability to perform licensed duties. It is therefore reportable to the Ops Training Manager using Data Sheet 1.

LESSON PLAN/OBJ: RO-C-ADM06/Terminal and #3
REFERENCE: OHI-2071

KA - 194001 2.1.4
Generic
Conduct of Operations
Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, 'no-solo' operation, maintenance of active license status, 10CFR55, etc.
RO - 3.3 SRO - 3.8
CFR - 41.10 / 43.2

KA Justification - Question tests knowledge of the operators responsibility for making notifications regarding changes to medical condition.

Original Question # - Master Bank OpMED-19
Original Question KA - Unknown

67. 067 004/BOTH/OK/DIRECT/NRC EXAM 2006-094-12/194001 2.1.5/2.9/3.9/F/2

Given the following conditions on Unit 2:

- Reactor power is stable at 100%.
- A Reactor Operator and the Unit Supervisor are in the Control Room.
- A high vibration alarm is received on the Heater Drain pump requiring someone to go behind the panel to check the indications.

Which ONE of the following describes the procedurally accepted method of checking the indications?

A.✓ The Unit Supervisor can go behind the panel to check the vibration.

B. The Reactor Operator can go behind the panel to check the vibration.

C. Both the Reactor Operator and the Unit Supervisor are allowed to go behind the panel to check the vibration as long as all controls are in automatic.

D. Neither the Reactor Operator or the Unit Supervisor can go behind the panels. They must get another operator to check the vibration.

ANSWER: A

A - CORRECT. The Unit Supervisor must be in the Control Room but may go behind the panels. The RO must remain in the view of the panels.
B - INCORRECT. The RO must remain in the view of the panels.
C - INCORRECT. The RO must remain in the view of the panels.
D - INCORRECT. The SRO may go behind the panels.

LESSON PLAN/OBJ: RO-C-ADM01/#1
REFERENCE: OHI- 4000, Conduct of Operations,
Attachment 22 (Shift Staffing)

KA - 194001 2.1.5
Generic
Conduct of Operations
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.
RO - 2.9 SRO - 3.9
CFR - 41.10 / 43.5 / 45.12

KA Justification - Question tests ability to understand and use procedures governing staffing and At-the-Controls areas.

Original Question # - Cook 2006 NRC Exam-094-12 : AUDIT RO22-BOTH-25
Original Question KA - 2.1.4

68. 068 001/BOTH/OK/NEW/NEW/194001 2.1.39/3.6/4.3/F/2

Conservative Decision making states that :

"**WHEN** faced with unexpected or uncertain conditions, **THEN** personnel must promptly identify a transition point at which efforts to keep the unit on-line or on schedule are no longer conservative, nor reasonable."

Once this point is reached :

A. the reactor must be tripped immediately.

B. senior management must be notified to determine course of action.

C.✓ actions must be taken to place the unit in a safe condition without hesitation.

D. the NRC must be notified and actions taken to address the problem within one hour.

ANSWER: C

A - INCORRECT. tripping the reactor is only one of the options available and may not be the prudent choice based on the transition point determined.
B - INCORRECT. Action must be taken without hesitation.
C - CORRECT. OHI-4000, Att. 5, Step 3.4 states "**WHEN** faced with unexpected or uncertain conditions, **THEN** personnel must promptly identify a transition point at which efforts to keep the unit on-line or on schedule are no longer conservative, nor reasonable. Once this point is reached, actions to place the unit in a safe condition by reducing power, tripping the reactor, or suspending core alterations must be taken without hesitation.
D - INCORRECT. NRC Notification is not required and action must be taken without hesitation.

LESSON PLAN/OBJ: RO-C-ADM14/#ADM14-3
REFERENCE: OHI-4000, Attachment 5

KA - 194001 2.1.39
Generic
Conduct of Operations
Knowledge of conservative decision making practices.
RO - 3.6 SRO - 4.3
CFR - 41.10 / 43.5 / 45.12

KA Justification - Question tests Knowledge of conservative decision making practices.

Original Question # - New
Original Question KA - New

69. 069 003/BOTH/OK/DIRECT/MAST 01ADMC0302-1/194001 2.2.20/2.6/3.8/F/3

In accordance with administrative procedures, which ONE of the following conditions would permit troubleshooting to be performed on Technical Specification equipment?

A. Equipment is operating.

B.✓ Equipment is declared inoperable.

C. Equipment is logged operable but degraded.

D. Equipment is NOT operating and redundant train is in operation.

ANSWER: B

A - INCORRECT. The fact that TS equipment is operating would not make it acceptable to perform trouble shooting activities.
B - CORRECT. Troubleshooting activities are allowed if the equipment is declared inoperable.
C - INCORRECT. Equipment may be logged as operable but degraded but this is not a factor in determining if troubleshooting can be performed.
D - INCORRECT. There is no provision to perform activities just because the equipment is not operating. The equipment must be inoperable or restored with an approved procedure.

PMP-2291-TRS-001:
3.1.5 Troubleshooting activities shall not be performed on Technical Specification equipment that results in the equipment becoming inoperable
-UNLESS-
The troubleshooting is performed in conjunction with an approved procedure that returns the equipment to an operable status,
AND
Permission is obtained from the Shift Manager.
3.1.6 Troubleshooting activities may be performed on Technical Specification equipment that is out of service or inoperable.

LESSON PLAN/OBJ: RO-C-ADM03/RO-C-ADM03-E2, RO-C-ADM03-E7
REFERENCE: PMP-2291-TRS-001, Step 3.1.5 & 3.1.6.

KA - 194001 2.2.20
Generic
Equipment Control
Knowledge of the process for managing troubleshooting activities.
RO - 2.6 SRO - 3.8
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Question tests knowledge of the troubleshooting process (when allowed on TS equipment).

Original Question # - MASTER 01ADMC0302-1
Original Question KA - 2.1.12

70. 070 002/BOTH/OK/DIRECT/NRC EXAM 2004-064-5/194001 2.2.21/2.9/4.1/F/3

A maintenance visual inspection requires momentarily placing a **'B' Train** pump control switch in PULL-TO-LOCKOUT. The Unit condition is such that BOTH trains are required to auto start.

Which ONE of the following describes the status of the affected ESF system?

The **'B' Train** pump is INOPERABLE until:

A. ✓ the control switch is independently verified in its normal position.

B. the pump's monthly surveillance has been performed.

C. the pump's auto start function is tested.

D. the pump is manually started.

ANSWER: A

A - CORRECT.	The B train pump may be considered OPERABLE after being returned to the correct position and being independently verified.
B - INCORRECT.	Surveillance does NOT need to be performed to declare B train equipment OPERABLE.
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 - INCORRECT.	Once returned to the correct position and being independently verified train B is considered OPERABLE - a functional test (manual start) is NOT required.

LESSON PLAN/OBJ: RO-O-0005/#ADM0170301
REFERENCE: OHI-4043, Technical specification Open Items Log

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

KA Justification - Requires knowledge of the requirements for declaring a pump OPERABLE following maintenance activity.

Original Question # - NRC EXAM 2004-064-5
Original Question KA - GENERIC 2.2.24

71. 071 002/BOTH/OK/DIRECT/RO26 AUDIT-98/194001 2.3.4/3.2/3.7/F/3

Given the following conditions on Unit 2:

- A LOCA has occurred
- The TSC has been fully staffed and activated
- An individual is needed for lifesaving activities during which 30 Rem of TEDE exposure is expected to be received

Which ONE of the following is correct concerning this lifesaving activity?

A.✓ The individual is required to be a volunteer and the Site Emergency Coordinator is required to approve the exposure.

B. The individual is required to be a volunteer and the Operations Shift Manager is required to approve the exposure.

C. The individual is NOT required to be a volunteer and the Site Emergency Coordinator is required to approve the exposure.

D. The individual is NOT required to be a volunteer and the Operations Shift Manager is required to approve the exposure.

ANSWER: A

A - CORRECT. Once in a lifetime doses in excess of 25 REM require a person to be a volunteer. Any extension above 10CFR20 limits requires SEC approval.

B - INCORRECT. The **SEC** approves extensions above 10CFR20 limits. (SM has been relieved of SEC duties since TSC is activated.

C - INCORRECT. A volunteer is required if the dose is in excess of 25 REM.

D - INCORRECT. The **SEC** approves extensions above 10CFR20 limits.(SM has been relieved of SEC duties since TSC is activated.

LESSON PLAN/OBJ: RO-C-RP02/#RO-C-RP02-E6
REFERENCE: RO-C-RP02, RMT-2080-TSC-001, Attachment 13

KA - 194001 2.3.4
Generic
Radiation Control
Knowledge of radiation exposure limits under normal or emergency conditions.
RO - 3.2 SRO - 3.7
CFR - 41.12 / 43.4 / 45.10
SCLR - 1P

K/A Justification - Requires knowledge of the conditions required to allow for an emergency dose during an accident.

Modified Question to raise dose level to 30 R. Changed correct Answer to A vs. C.
Original Question - RO25 Audit-93, CATAWBA2005, RO26 Audit-98

72. 072 003/BOTH/OK/DIRECT/NRC EXAM 2004-099-3/194001 2.3.7/3.5/3.6/H/3

Given the following conditions:

- Units 1 and 2 are at 100% power.
- Unit 2 has experienced several fuel pin failures.
- A leak must be repaired on a pipe at the end of the Aux. Bldg. 601 ft. elev. pipe tunnel.
- The general area dose rate in the location of the repair is 600 mrem/hr.
- In order to reach the location of the repair the worker must transit through a 6 Rem/hr high radiation area for 2 minutes and return via the same path.
- The worker currently has an accumulated annual dose of 400 mrem.

Which ONE of the following is the maximum allowable time that the worker can participate in the repairs and NOT exceed the TEDE Administrative Dose Limit?

A. 70 minutes

B.✓ 120 minutes

C. 140 minutes

D. 160 minutes

ANSWER: B

A - INCORRECT. Based on using a limit of 1500 versus correct ADL (2000).
B - CORRECT. The candidate should determine that the ADL is 2000 mrem. Transient exposure is 400 mrem (6000mrem/hr x 4/60hr). (transit to and from the job). (Current) 400 mrem + (transit) 400 mrem = 800 mrem ADL of 2000 mrem - 800 mrem = 1200 mrem allowable before reaching ADL.
1200 mrem /600 mrem/hr = 2 hours
C - INCORRECT. Based on calculating using a one-way transit dose.
D - INCORRECT. Based on using ADL (2000) and NO transit dose.

LESSON PLAN/OBJ: RO-C-RP02/#5
REFERENCE: RO-C-RO02 10CFR20 and Radiation Protection
Attachment pg. 1

KA - 194001 2.3.7
Generic
Radiation Control
Ability to comply with radiation work permit requirements during normal or abnormal conditions.
RO - 3.5 SRO - 3.6
CFR - 41.12 / 45.10

KA Justification - Question tests knowledge of stay time, transition, time, and dose limits required for compliance with an RWP.

Original Question # - NRC Exam 2004-099-3, AUDIT02-SRO-6
Original Question KA - GENERIC 2.3.1

73. 073 002/BOTH/OK/DIRECT/NRC EXAM 2004-049-5/194001 2.3.11/3.8/4.3/H/3

Given the following conditions on Unit 1:

- Containment Purge System is operating in the VENTILATION MODE.
- A HIGH alarm on VRS-1505, Auxiliary Building Ventilation Noble Gas Activity Monitor, occurs (unplanned).

Which ONE of the following describes the required operator response for the Containment Ventilation System to the failure alarm?

A.✓ Stop the Containment Purge and consult with Radiation Protection prior to restarting the system.

B. Continue the Purge as long as VRS-1101, Containment Normal Range Area Radiation Monitor still indicating as expected.

C. Verify the following:
- 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.

D. Verify the following:
- 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: A

A - CORRECT. When the Containment Purge system is operating in the Ventilation Mode, the automatic isolation signals are blocked. The procedure requires the Purge to be stopped and radiation protection concurrence prior to restarting the system.

B - INCORRECT. The procedure requires the Purge to be stopped and radiation protection notified. Plausible as the Containment radiation monitor is still operable monitoring for any release

C - INCORRECT. When the Containment Purge system is operating in the Ventilation Mode, the automatic isolation signals are blocked. Plausible as these are functions from containment isolation signal actuation results.

D - INCORRECT. When the Containment Purge system is operating in the Ventilation Mode, the automatic isolation signals are blocked. Plausible as these are functions from containment isolation signal actuation results.

LESSON PLAN/OBJ: RO-C-02800/#9
REFERENCE: 1-OHP-4021-028-005, Att. 2

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

KA Justification - Question tests ability to control releases through identification of conditions requiring manual termination.

Original Question # - NRC Exam 2004-049-5, AUDIT02-BOTH31
Original Question KA - GENERIC 2.3.9

74. 074 004/BOTH/OK/DIRECT/22430-DIAB2002-63/194001 2.4.2/4.5/4.6/F/3

Given the following conditions on Unit 1:

- A reactor trip occurred from 15% power.
- Safety Injection was NOT actuated and was NOT required.
- 1-OHP-4023-E-0, Reactor Trip or Safety Injection, has been performed, and a transition to 1-OHP-4023-ES-0.1, Reactor Trip Response, has been made.

The following conditions exist:

- Tavg is STABLE at 547°F
- Pressurizer level is 11% and lowering slowly
- RCS subcooling is 32°F and lowering slowly
- All NR SG levels are 28 - 30%;
- AFW flows indicate 0 klb/hr
- Containment pressure is 0.7 psig and rising slowly

Which ONE of the following describes the appropriate actions for these conditions?

A.✓ Actuate SI and return to 1-OHP-4023-E-0 step 1.

B. Go to 1-OHP-4023-FR-H.1, Response to Loss of Secondary Heat Sink.

C. Go to 1-OHP-4023-FR-I.2, Response to Low Pressurizer Level.

D. Manually start ECCS pumps and continue with 1-OHP-4023-ES-0.1.

ANSWER: A

A - CORRECT. The foldout page directs an SI and return to 1-OHP-4023-E-0 if subcooling is < 40°F

B - INCORRECT. SG levels are adequate for Heat Sink even with no AFW flow. Plausible due to loss of subcooling the student could want to increase cooling to the S/G as well as missing the Narrow Range level.

C - INCORRECT. Initiating SI and returning to 1-OHP-4023-E-0 would take precedence over 1-OHP-4023-FR-I.2. Plausible due to the desire of the student to recover PZR level.

D - INCORRECT. This may be the action required in other emergency procedures (ES-1.2) but a return to 1-OHP-4023-E-0 is required to verify proper alignment of ECCS equipment. ES-0.1 is considered a 'non-accident' EOP. Plausible due to this action being elsewhere in the EOP network.

LESSON PLAN/OBJ: RO-C-EOP03/#18 & 23
REFERENCE: 1-OHP-4021-ES-0.1 (Foldout Page)

KA - 194001 2.4.2
Generic
Emergency Procedures/Plan
Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.
RO - 4.5 SRO - 4.6
CFR - 41.7 / 45.7 / 45.8

KA Justification - Requires a knowledge of conditions(setpoints) that would require a transition from Reactor Trip Response to re-entry into the Reactor Trip/SI EOP.

Original Question # - 22430-DIAB2002-63
Original Question KA - E05.2.4.21

75. 075 002/BOTH/OK/DIRECT/RO22 AUDIT-BOTH-13/194001 2.4.6/3.7/4.7/F/3

Given the following conditions on Unit 2:

- A Safety Injection (SI) has ocurred.
- The Immediate Action steps of 2-OHP-4023-E-0, Reactor Trip Or Safety Injection, have just been completed.

<u>The following steam generator conditions are noted:</u>

- SG 21 pressure is 740 psig and lowering slowly.
- SG 22 pressure is 450 psig and lowering rapidly
- SG 23 pressure is 735 psig and lowering slowly.
- SG 24 pressure is 745 psig and lowering slowly.
- Main Steam header pressure is 700 psig and lowering slowly.

Which ONE of the following actions should be promptly performed to mitigate the event?

A. Transition to 2-OHP-4023-E-1, Loss of Reactor or Secondary Coolant.

B. Transition to 2-OHP-4023-E-2, Faulted Steam Generator Isolation,

C.✓ Close all the SG stop valves and continue with 2-OHP-4023-E-0, Reactor Trip Or Safety Injection.

D. Close SG 22 Stop Valve and verify steam supply available to the Turbine Driven Auxiliary Feedwater Pump (TDAFP)
.

ANSWER: C

A - INCORRECT. Step transition to E-1 is not performed until later in procedures. In addition, conditions given indicate a steam leak (E-2) condition. Plausible due to E-0 does direct going to E-1.

B - INCORRECT. Plausible due to conditions described indicate a Faulted S/G but the transition is not allowed at this point in E-0. The transition to E-2 is later in the procedure.

C - CORRECT. Steamlines should be isolated for any number of reasons, such as Automatic steamline isolation failure, RCS temperature lowering (procedural steps)and personnel protection. OHI-4023 allows prudent actions to trip the SG Stop Valves when it is apparent that a steam line leak has occured for personnel protection and in response to automatic action failures.

D - INCORRECT. All SG Stop valves should be closed when taking the prudent actions in OHI-4023. Plausible due to student diagnosing the problem and taking actions only to isolate the affected S/G.

LESSON PLAN/OBJ: RO-C-EOP01/#17, RO-C-EOP07/#3
REFERENCE: OHI-4023, Abnormal/Emergency Procedure User's Guide, Step 4.7.3.b.5.

KA - 194001 2.4.6
Generic
Emergency Procedures/Plan
Knowledge of EOP mitigation strategies.
RO - 3.7 SRO - 4.7
CFR - 41.10 / 43.5 / 45.13

KA Justification - Requires the knowledge of the mitigating strategy for limiting the cooldown in the EOPs during a Steam Line break before the diagnostic steps have been reached (OHI-4023 allowance to close SG Stop Valves).

Original Question # - RO22 Audit-Both-13
Original Question KA - APE 040 AK3.04

76. 076 005/SRO/OK/MODIFIED/RO26 AUDIT-95/000005 AA2.03/3.5/4.4/H/3

Given the following conditions on Unit 2:

- Unit was at 90% power
- Control Rods were in AUTO
- CBD began to step out with no mismatch signal.
- Rods were taken to MANUAL and rod motion ceased.

The following conditions now exist:

- CBD Bank Demand position is now 222 Steps.
- Group 2 Bank D RPIs ALL indicate 222 Steps.
- Group 1 Bank D RPIs indicate as follows:
  - Rod D4: 205 Steps.
  - Rod D12: 222 Steps
  - Rod M12: 207 Steps
  - Rod M4: 222 Steps

Reactor Engineering has determined that all CBD rods are free to move and has provided the following information:

- R is 1.041
- CFQ is 2.335
- K(Z) is .95 (at 10 feet)
- $F_Q^W(Z)$ is 2.174 (at 10 feet)

Which ONE of the following identifies the Technical Specification Action Condition(s) that must be entered?

Reference Provided

A. 3.1.4.A Only

B. 3.1.4.B Only

C. 3.1.4.A and 3.1.4.B Only

D.✓ 3.1.4.B and 3.1.4.D Only

ANSWER: D

A - INCORRECT. Rods are free to move, as stated in the stem, thus they are all operable.
B - INCORRECT. Using the numbers provided with Figure 3.1.4-1, rods which are <14 steps misaligned meet alignment requirements. This is the correct action with only 1 misaligned rod.
C - INCORRECT. Rods are free to move and are operable.
D - CORRECT. Both D4 and M12 are unacceptably misaligned. D4 is 17 steps off and M12 is 15 steps off.

LESSON PLAN/OBJ: RO-C-TS01/#11
REFERENCE: TS 3.1.4, Rod Group Alignment Limits

Reference Provided: Unit 2 TS 3.1.4 Rod Group Alignment Limits

MODIFIED: Changed to have 2 rods misaligned. Changed answer to D from B.

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

KA Justification - Requires the ability to determine and interpret the required TS actions with 2 misaligned rods.

Original Question # - RO26 AUDIT-95
Original Question KA - 194001 2.1.7

77. 077 008/SRO/OK/NEW/NEW/000007 EA2.04/4.6/4.4/H/3

Given the following conditions on Unit 1:

- A power escalation is in progress
- At approximately 35% power the Main Turbine Trips

The following conditions now exist:

- All 4 Turbine Stop Valve Status Lights are lit
- All RPIs have lost power
- Train A Reactor Trip Breaker is Open
- Train B Reactor Trip Breaker is Closed
- WR Log Power = 7%
- WR Startup Rate = 0.0 DPM and stable

Which ONE of the following actions is required?

A. Implement 1-OHP-4023-E-0, Reactor Trip or Safety Injection.

Following completion of Immediate Actions, transition to 1-OHP-4023-ES-0.1, Reactor Trip Response.

B.✓ Implement 1-OHP-4023-E-0, Reactor Trip or Safety Injection.

During verification of Reactor Trip, transition to Implement 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS, and manually insert control rods.

C. Implement 1-OHP-4022-001-002, Loss of Load (Load Rejection).

When directed go to 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS.

D. Implement 1-OHP-4022-001-002, Loss of Load (Load Rejection).

Upon Turbine Trip Verification, go to 1-OHP-4023-E.0, Reactor Trip or Safety Injection.

ANSWER: B

A - INCORRECT. A transition to FR-S.1 is required. Plausible as E-0 is the correct entry procedure and student may consider actions in E-0 capable of resolving the problem.

B - CORRECT. Due to flux being greater than 5% and not lowering, E-0, Step 1 RNO will require a transition to FR-S.1. Since Reactor trip cannot be verified (flux NOT lowering), manual control rod insertion will be required.

C - INCORRECT. E-0 should be implemented since a Turbine Trip greater than P-8 should result in a Reactor trip. Plausible as student may key in on current power level and go to procedure governing for a loss of load with a reactor power level below the reactor trip setpoint.

D - INCORRECT. E-0 should be implemented since a Turbine Trip greater than P-8 should result in a Reactor trip. E-0 takes precedence over AOPs. Plausible as student may key in on current power level and go to procedure governing for a loss of load with a reactor power level below the reactor trip setpoint.

LESSON PLAN/OBJ: RO-C-EOP03/#13, #14, RO-C-EOP04/#13
REFERENCE: 1-OHP-4023-E-0, Reactor Trip or Safety Injection, 1-OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS

KA - 000007 EA2.04
Reactor Trip - Stabilization
Ability to determine and interpret the following as they apply to a reactor trip:
If reactor should have tripped but has not done so, manually trip the reactor and carry out actions in ATWS EOP
RO - 4.6 SRO - 4.4
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Requires the ability to determine that a Reactor Trip is required and that the reactor is not tripped (FR-S.1 Entry), and then the immediate actions needed to make the reactor subcritical (inserting control rods).

Original Question # - RO23 AUDIT 011-4

78. 078 003/SRO/OK/MODIFIED/NRC EXAM 2004-020-2/000022 2.4.50/4.2/4.0/H/3

Given the following conditions on Unit 2:

- Reactor power was 100% power when an RCS leak developed.
- The Unit Supervisor is implementing 2-OHP 4022-002-020, Excessive RCS Leakage.

The following conditions now exist:

- Letdown flow is isolated.
- East and West Charging pumps are operating.
- Charging flow is 180 gpm.
- Pressurizer level is 51% and constant.
- VCT makeup is in Auto.
- VCT level is 22% and lowering.
- Containment pressure is 0.5 psig and constant.

Which ONE of the following describes the required operator action and why (assume all control systems function as designed)?

A. Verify that CCP suction automatically aligns to the RWST at 14.0% VCT level and perform a controlled rapid shutdown per 2-OHP-4022-001-006 Rapid Power Reduction Response, to maintain RCS Tavg-Tref.

B. Verify VCT auto makeup begins at 14.0 % and then restore 75 gpm letdown to ensure proper regen heat exchanger warming of the charging flow.

C. Verify that CCP suction automatically aligns to the RWST at 7.0% VCT level and perform a controlled rapid shutdown per 2-OHP-4022-001-006 Rapid Power Reduction Response since RCS leakage is greater than the Technical Specification Limit.

D✓ Trip the reactor and transition to 2-OHP-4023-E-0, Reactor Trip or Safety Injection since VCT level can NOT be maintained with VCT auto makeup.

ANSWER: D

A - INCORRECT. The procedure directs a Reactor Trip. Temperature control would be extremely difficult. The VCT Refueling Water Sequence Actuation and alarm come in at 2.5% VCT level. Plausible since VCT low alarm setpoint is 14.0% and RWST alignment would impact temperature control.

B - INCORRECT. Letdown was isolated to allow Pressurizer level to be stabilized. Auto Makeup will start at 24% but flow will not be sufficient to make up for 180 gpm leak.

C - INCORRECT. The procedure directs a Reactor Trip. The VCT Refueling Water Sequence Actuation and alarm come in at 2.5% VCT level. Plausible since a VCT low alarm setpoint is at 7.0% and this leakage would exceed TS.

D - CORRECT. Leakage in excess of VCT makeup will lead to eventual loss of CCP suction. This would be mitigated by the refueling water sequence swapover to the RWST suction source but this would result in excessive boration of the RCS. Lowering level in excess of auto makeup capability require a RX trip per 2-OHP-4022-002-020.

LESSON PLAN/OBJ: RO-C-AOP-D1/RO-C-AOP0160412-E3
REFERENCE: 2-OHP-4022-002-020, Excessive Reactor Coolant Leakage Step 3, 2-OHP-4024-209, Drops 48, 49, 50

KA - 000022 2.4.50
Loss of Reactor Coolant Makeup
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

KA Justification - Requires the ability to determine that with the stated conditions VCT Makeup will be insufficient to maintain VCT level. Further lowering of VCT level will eventually lead to the Refueling Water Sequence alarm setpoint and actuation. Based on these conditions, the SRO is required to assess the plant conditions and determine that a reactor trip and entry into the EOPs is required.

Modified by changing VCT Makeup in the Stem (was operating at maximum) to in Auto. Changed distractors A & C to add RWST swapover setpoints. and B & D to include VCT Makeup occurring or not.

Original Question # - NRC 2004-020-2
Original Question KA - 002000 A2.01

79. 079 002/SRO/OK/MODIFIED/NRC EXAM 2008-77/000026 AA2.02/2.9/3.6/H/3

Given the following conditions on Unit 2:

- Unit was operating at 100% power
- 2-OHP-4022-016-001, Malfunction of the CCW System is being implemented due to indications of a lowering CCW Surge tank level.
- The Crew has started the West CCW pump, split the East and West Headers, and aligned the Miscellaneous Services Header to the East Header.
- An AEO has reported that a CCW leak of approximately 150 gpm has been identified in the Aux Building 609' elevation, flowing toward the passenger elevator.

The following Surge Tank Level Recorder conditions exist:

	Train 'A' CLR-410	**Train 'B' CLR-411**
Reading	48"	18"
Trend	Stable	Lowering

The leak is located on the_____(1)______ Header. The Unit Supervisor will direct the crew to ____(2)______.

A. (1) Miscellaneous Services
(2) 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. (1) East Safeguards
(2) Shutdown the East CCW pump and align the Miscellaneous Services Header to the West Safeguards Header in accordance with 2-OHP-4022-016-001, Malfunction of the CCW System.

C.✓ (1) West Safeguards
(2) Shutdown the West CCW pump and the equipment cooled by the West Header in accordance with 2-OHP-4022-016-001, Malfunction of the CCW System.

D. (1) Miscellaneous Services
(2) 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: C

A - INCORRECT. During the initial train split, the Misc. Header is aligned to the East Safeguards Header. Miscellaneous Header is isolated from the leak. Plausible if the student believes the leak is on the Misc header. Differs from distractor D as to the additional actions listed. This answer stops the CCW pumps and implements loss of CCW while performing E-0 which is logical for mitigating the event however it is NOT in full compliance with the procedural requirements.

B - INCORRECT. During the initial train split, the Misc. Header is aligned to the East Safeguards Header. The East CCW Header is isolated from the leak. Plausible if the student determines the leak is on the incorrect header.

C - CORRECT. Initial train separation places the Miscellaneous Header on the East Train. The conditions presented indicate that the leak is on the West Safeguards Header which can be isolated from the East Header amd Miscellaneous Header.

D - INCORRECT. Miscellaneous Header is aligned to the East CCW Train. Miscellaneous Header does not need to be isolated. Plausible if the student believes the leak is on the Misc header. Differs from distractor A as to the additional actions listed. This answer trips the reactor, RCP’s, and isolates the Misc header while performing E-0 which is logical for mitigating the event however it is NOT in full compliance with the procedural requirements.

LESSON PLAN/OBJ: RO-C-AOP-D8\RO-C-AOP0420412-E3, RO-C-01600-E3
REFERENCE: 2-OHP-4022-016-001, Malfunction of the CCW System, SOD-01600-001, RO-C-AOP-D8

Modified: Stem surge tank levels (Swapped) and leakage location. This changed the correct answer from D to C.

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

KA Justification - Question requires Candidate to determine the leak location and determine the correct procedural actions required based on the assessment of plant conditions.

Original Question # - NRC EXAM 2008-77
Original Question KA - 000026 AA2.02

80. 080 004/SRO/OK/DIRECT/RO24 AUDIT-021-7/000032 2.1.7/4.4/4.7/H/3

Given the following conditions on Unit 2:

- A reactor startup is in progress with the reactor just critical.
- The operator has just stopped moving control rods.
- Intermediate Range Power slowly rises above 2 x 10^{-10} amps.
- ONE source range (SR) nuclear instrumentation channel (N-31) fails LOW.
- Remaining power indications stabilize.

Which ONE of the following actions, if any, is required for compliance with Technical Specifications?

Reference Provided

A.✓ No action required, source range not required to be operable.

B. Trip the reactor and enter 2-OHP-4023-E-0, Reactor Trip or Safety Injection.

C. Conduct a reactor shutdown and restore both SR channels to operability prior to next startup.

D. Suspend all operations involving positive reactivity changes until both SR channels are restored to operability.

ANSWER: A

A - CORRECT. TS 3.3.1 (Instrumentation) establishes that above P-6, the SR NI are not required by TS and will shortly be de-energized by procedure. Since there are no TS implications, the startup may proceed.

B - INCORRECT. No entry conditions are met for a reactor trip. (plausible misconception) Plausible as this would be a conservative action to place the plant in a known safe, stable condition but does not answer the actual question.

C - INCORRECT. A plant shutdown is not required unless both SR channels are lost during reactor startup. Plausible since shutdown is performed for several startup inconsistencies (ECC wrong, conditions change, etc.) and this is an action for several reactor start up issues.

D - INCORRECT. SR channels can be blocked. Reactor power is above P-6. Plausible since this statement would be correct if reactor power was < P-6 and this is an action associated with the Source Range Detectors in Technical Specifications.

LESSON PLAN/OBJ: RO-C-01300/#21
REFERENCE: TS 3.3.1, RTS Instrumentation, Table 3.3.1-1

Reference Provided: Unit 2 TS 3.3.1, Reactor Trip System Instrumentation

KA - 000032 2.1.7
Loss of Source Range Nuclear Instrumentation
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

KA Justification - Requires an evaluation of the plant status due to a loss of source rage instruments above P-6 setpoint and to determine per TS any actions required.

Original Question # - RO24 AUDIT-021-7
Original Question KA - APE.032GEN2.1.20

81. 081 010/SRO/OK/NEW/NEW/000038 EA2.09/4.2/4.2/H/3

Given the following conditions on Unit 2:

- A SGTR has occurred coincident with a Loss of Offsite Power
- 2-OHP-4023-E-3, Steam Generator Tube Rupture, is being performed.
- The Unit Supervisor is at Step 41, Select Appropriate Post-SGTR Cooldown Procedure.

The following conditions now exist:

- RCS Wide Range Pressure is 800 psig and stable.
- RCS Incore Thermocouples are 475°F and slowly lowering
- RCS T-hots are 470 and slowly lowering
- RCS T-colds are 445 and stable
- SG Pressures are 387 psig and stable.
- PZR level is 25% and slowly rising.

Which ONE of the following describes the status of natural circulation and the appropriate procedural transition for the Unit Supervisor?

A.✓ Natural Circulation exists.
Transition to 2-OHP-4023-ES-3.1, Post-SGTR Cooldown Using Backfill.

B. Natural Circulation does NOT exist.
Transition to 2-OHP-4023-ES-3.2, Post-SGTR Cooldown Using Blowdown

C. Natural Circulation does NOT exist.
Transition to 2-OHP-4023-ES-3.3, Post-SGTR Cooldown Using Steam Dump.

D. Natural Circulation exists.
Transition to 2-OHP-4023-ECA-3.1, SGTR With Loss Of Reactor Coolant - Subcooled Recovery Desired.

ANSWER: A

A - CORRECT. Based on the conditions Natural Circ exists: T/Cs, T-hot, and SG Press. (Stable or Lowering); RCS cold leg temperatures are at saturation Temp for SG Press; Subcooling is 51ºF (>40ºF). 2-OHP-4023-ES-3.1, Post-SGTR Cooldown Using Backfill, is an appropriate transition form E-3.

B - INCORRECT. Based on the conditions Natural Circ DOES exist. Plausible as this procedure is a valid transition from E-3 but incorrect if the student can not validate the status of natural circulation.

C - INCORRECT. Based on the conditions Natural Circ DOES exist. Plausible as this procedure is a valid transition from E-3 but incorrect if the student can not validate the status of natural circulation.

D - INCORRECT. ECA-3.1, SGTR With Loss Of Reactor Coolant - Subcooled Recovery Desired, transition is from the foldout page when either subcooling or PZR Level cannot be maintained. PZR level and subcooling are both adequate. Plausible since Natural Circulation requires subcooling and student knows that cooldown of the ruptured SG will be somewhat impeded under natural circulation conditions.

LESSON PLAN/OBJ: RO-C-EOP08/#15, #22
REFERENCE: 2-OHP-4023-E-3, Steam Generator Tube Rupture

KA - 000038 EA2.09
Steam Generator Tube Rupture (SGTR)
Ability to determine and interpret the following as they apply to a SGTR:
Existence of natural circulation, using plant parameters
RO - 4.2 SRO - 4.2
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Requires knowledge of the conditions that indicate natural circ during a SGTR event and (SRO) the appropriate transition based on that determination.

Original Question # - New
Original Question KA - New

82. 082 003/SRO/OK/DIRECT/NRC EXAM 2006-080-20/000040 2.4.4/4.5/4.7/H/3

Given the following conditions on Unit 2:

- Steam Gen 1/2/3/4 Steam Line Flow High Alarms - LIT
- Steam Gen 1/2/3/4 SF>FWF Flow Mismatch Alarms - LIT
- RCS Tavg is 561°F and lowering
- Turbine load is lowering
- Rods are stepping out
- Steam flows are 3.6 x 10^6 lbm/hr and stable.
- FW flows are 2.1 x 10^6 lbm/hr and rising.

Which ONE of the following correctly describes the cause and required action to be taken for the above conditions?

A.✓ A steam line break exists. Direct the operators to perform a Reactor Trip and Main Steamline Isolation.

B. A feed line break exists. Direct the operators to perform a Reactor Trip and Main Feedwater Isolation.

C. Feedwater Pump Delta-P is too Low. Direct the operator to raise FW Pump Speed and FW pump flow.

D. MPC-253 has failed LOW. Direct the operators to perform actions for failed First Stage Turbine Impulse Pressure Transmitter.

ANSWER: A

A - CORRECT. Based on the conditions presented a steam line break has occurred. Steam flow is indicating at the 97 to 98% power range. Tavg is 13°F Low for 98% power. A reactor trip and Steam Line isolation is warranted.

B - INCORRECT. If a FW break existed RCS temperature would be rising. Plausible as the alarms are associated with Feed and well as steam.

C - INCORRECT. If FW Flow was low RCS Temperature would be rising. Plausible as feed is less than steam which can be caused by SGFP low DP.

D - INCORRECT. If MPC-253 failed low the alarms would come in (Steam flow higher than calculated power) but rods would step out and SF/FWF mismatch would not be this high. Plausible as this failure would drive some of the alarms being received.

LESSON PLAN/OBJ: RO-C-EOP07/#5
REFERENCE: RO-C-EOP07

KA - 000040 2.4.4
Steam Line Rupture
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

KA Justification - Requires knowledge of the conditions that would require entry into the reactor trip procedure and the actions needed to address the steamline break (Main Steam Isolation)

Original Question # - Cook 2006 NRC Exam-080-20, Modified from NRC EXAM 2004-073-1
Original Question KA - 000040 AA2.01

83. 083 007/SRO/OK/NEW/NEW/000060 2.4.45/2.9/3.1/H/3

Given the following conditions:

- A Gas Decay Tank release is in progress through the Unit 1 plant vent.
- A High Alarm occurs on VRS-2505, Unit 2 Vent Low Range Noble Gas Radiation Monitor.
- RP determines that VRS-2505 is INOPERABLE due to a failed high channel.

The effects of the VRS-2505 High Alarm are that 12-RRV-306, Vent Stack Release Valve, __________________.

Additionally, which ONE of the following is required by PMP-6010-OSD-001, Off-site Dose Calculation Manual?

Reference Provided

A.✓ will automatically close

Actions 6 and 9

B. must be manually closed

Actions 6 and 9

C. will automatically close

Action 9 Only

D. must be manually closed

Action 6 Only

ANSWER: A

A - CORRECT. A high alarm on VRS-2505 will automatically close 12-RRV-306. PMP-6010-OSD-001, Att. 3.4, Action 6 requires grab samples to be taken at least once per shift and analyzed for gross activity within 24 hours for continued effluent release through the vent header.

B - INCORRECT. A high alarm on VRS-2505 will automatically close 12-RRV-306. Plausible as the ODCM requirements are correct.

C - INCORRECT. With VRS-2505 INOPERABLE, the ONE channel that monitors the Unit 2 Plant Vent is INOPERABLE. In accordance with PMP-6010-OSD-001, Att. 3.4, Action 6 is required at all times. Plausible as action 9 provides requirements for a batch release and the automatic closure of 12-RRV-306 is correct.

D - INCORRECT. A high alarm on VRS-2505 will automatically close 12-RRV-306. With VRS-2505 INOPERABLE, the ONE channel that monitors the Unit 2 Plant Vent is INOPERABLE. In accordance with PMP-6010-OSD-001, Att. 3.4, Action 6 is required at all times. Plausible as action 9 provides requirements for a batch release.

LESSON PLAN/OBJ: RO-C-02300/#3
REFERENCE: PMP-6010-OSD-001, Off-site Dose Calculation Manual, Attachment 3.4 (pages 57-59), 12-OHP-4024-139, Drop 5

Reference Provided: PMP-6010-OSD-001, Off-site Dose Calculation Manual, Attachment 3.4 (pages 57-59)

KA - 000060 2.4.45
Accidental Gaseous Radwaste Release
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

KA Justification - Requires the ability to determine the actions that are required for an alarm on vent stack rad monitor and the ability to use the Off-site Dose Calculation Manual, to determine the appropriate response to the rad monitor alarm.

Original Question # - New
Original Question KA - New

84. 084 008/SRO/OK/NEW/NEW/000062 2.4.20/3.8/4.3/F/3

Given the following conditions on Unit 2:

- Unit is implementing 2-OHP-4023-ECA-0.0, Loss of all AC Power
- Power has just been restored from the U2 AB EDG
- Cooling flow to the U2 AB EDG is being checked per Step 32
- **ONLY** the normal ESW supply to the U2 AB EDG is open and providing flow.

Which ONE of the following describes the actions the Unit Supervisor should direct and the reason for those actions?

A. Open the Alternate Supply to the AB EDG

Maximizes flow through the EDG to compensate for maximum loading.

B.✓ Leave the Alternate Supply to the AB EDG closed

Prevents a loss of ESW cooling to both trains of equipment due to silt and mud build-up in the component's heat exchangers.

C. Open the Alternate Supply to the AB EDG

Ensures adequate flow in the event of loss of the normal supply path.

D. Leave the Alternate Supply to the AB EDG closed

Limits the amount of flow to the EDG to ensure that other components in the ESW train receive adequate flow to support safety functions.

ANSWER: B

A - INCORRECT. Alternate supply should remain closed. See Answer B for justification. Plausible based on known loading for coming out of Loss of all AC with only one EDG.

B - CORRECT. Step 32 Note states "The alternate ESW cooling supply to the EDGs should remain isolated unless an EDG is running AND the normal ESW supply is NOT available." This Note was added to prevent a loss of ESW cooling from occurring to both trains of equipment due to slit and mud build-up in the component's heat exchangers if the ESW system trains are cross-tied via the alternate cooling supplies to the EDGs.

C - INCORRECT. Alternate supply should remain closed. See Answer B for justification. Plausible to ensure cooling to the only operating EDG for the unit.

D - INCORRECT. See Answer B for justification. Plausible based on ensuring cooling is provided to other required equipment.

LESSON PLAN/OBJ: RO-C-EOP14/#11
REFERENCE: 12-OHP-4023-ECA-0.0, EOP Step 32 (ERG Step N/A) Note 1 Background

KA - 000062 2.4.20
Loss of Nuclear Service Water
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

KA Justification - SRO Only portion is the application of the note prior to step 32 to the implementation of the step. Requires SRO to determine correct procedural actions required concerning ESW supply to the EDGs and the associated reason (operational implications) of the EOP note.

Original Question # - New
Original Question KA - New

85. 085 003/SRO/OK/DIRECT/CM-7746-38403/000069 AA2.01/3.7/4.3/H/3

Given the following conditions on Unit 1:

- Unit is preparing for a reactor start up following a refueling outage.
- Tavg is 515°F with a heatup in progress.
- During the outage testing was performed per the Containment Leak Rate Testing Program
- At 0200, a Station Engineer reports that a mistake had been made in analyzing the required Containment Leak Rate Test results that were conducted prior to exceeding 200°F.
- The initial calculated Type A leakage had been recorded as ACCEPTABLE
- Re-calculation indicates that the Type A leakage is actually UNACCEPTABLE
- The re-calculated values have been verified and reviewed by the Shift Manager

Which ONE of the following actions, if any, is required by Technical Specifications in response to this situation?

Reference Provided

A. Continue with the heatup. Do not enter Mode 2 until the leak test is re-performed

B. Enter Tech Spec 3.6.1. Remain in MODE 3 until performance of a risk assessment in accordance with Tech Spec 3.0.4.

C. Enter Tech Spec 3.0.3. Be in MODE 5 within 37 hours.

D✓ Enter Tech Spec 3.6.1. Be in MODE 5 within 37 hours.

ANSWER: D

A - INCORRECT. TS 3.6.1 is Applicable in Modes 1-4. Must comply with actions. Plausible if the student applies Mode change restraints to ensuring all required components are operable prior to changing Modes.

B - INCORRECT. TS 3.6.1 is Applicable in Modes 1-4. Must comply with actions including cooldown. Plausible if the student applies Mode change restraints to ensuring all required components are operable or a risk assessment has been performed prior to changing Modes (even lower mode entry).

C - INCORRECT. This is not a TS 3.0.3 issue; actions are provided in TS 3.6.1. Plausible if the student can not correctly apply tech Spec's but knows something is not correct

D - CORRECT. TS 3.6.1 required the Containment to be OPERABLE in Modes 1 - 4. If the Containment is INOPERABLE, then return to OPERABLE status in 1 hour or be in Mode 5 in the following 36 hours (37 hours to Mode 5)

LESSON PLAN/OBJ: RO-C-TS01/#11
REFERENCE: TS 3.6.1, Containment, Section 3.0

Reference Provided: Unit 1 TS 3.6.1, Containment, Section 3.0

KA - 000069 AA2.01
Loss of Containment Integrity
Ability to determine and interpret the following as they apply to the Loss of Containment Integrity:
Loss of containment integrity
RO - 3.7 SRO - 4.3
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Requires the ability to determine actions (in accordance with TS) when containment integrity is lost.

Original Question # - CM-7746-38403
Original Question KA - SYS 103 K3.02

86. 086 003/SRO/OK/DIRECT-REPEAT/NRC EXAM 2007-094/005000 2.4.21/4.0/4.6/F/3

Given the following conditions on Unit 1:

- A LOCA occurred 30 minutes ago
- RCS pressure is 125 psig
- RCS Core Exit TCs read 380°F
- RCS Cold Leg temperatures are 250°F
- 1N SI Pump is running providing 650 gpm flow
- 1E RHR Pump is running providing 3000 gpm flow

What is the appropriate action taken in response to the above conditions?

Entry into 1-OHP-4023-FR-P.1, Response to Pressurized Thermal Shock Condition, is:

A.✔ made but NO actions are implemented before returning to procedure in effect.

B. made and cooldown will continue within a limit of 50°F in any 60 minute period.

C. made and a RCS temperature soak for a ONE hour period will be completed.

D. NOT required since RCS pressure is below 300 psig.

ANSWER: A

A - CORRECT.	Entry into FR-P.1 is required due to the Orange Path with RCS at <285°F. The first step of P.1 checks RCS pressure at greater than 300 psig. Since Pressure is less than 300 psig and RHR flow is >400 gpm, no actions are performed and the operator is directed back to the procedure & step in effect.
B - INCORRECT.	Cooldown is not limited since the RCS has already experienced a large break. Plausible if the student does not recognize the exit criteria for the Red path with a LOCA.
C - INCORRECT.	A soak is not required since the RCS has already experienced a large break. Plausible if the student remembers the Red path has a soak requirement.
D - INCORRECT.	Entry into the procedure is still required and a pressure and flow check is made within the procedure. Plausible if the student knows the Red Path does not get implemented but incorrect in that the transition must still be made.

LESSON PLAN/OBJ: RO-C-EOP12/#29
REFERENCE: 1-OHP-4023-FR-P.1

KA - 005000 2.4.21
Residual Heat Removal System (RHRS)
Emergency Procedures/Plan
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.
RO - 4.0 SRO - 4.6
CFR - 41.7 / 43.5 / 45.12

KA Justification - Requires the ability to assess the status of PTS and recognize that for a LB LOCA (RHR Flow >400 gpm), that implementation of FR-P.1 is not required.

Original Question # - Cook NRC Exam 2007-094, INPO # 19406 Kewaunee, Unit 1 - 12/11/2000, RO26 Audit-80
Original Question KA - KA - 000011 EA2.14

87. 087 010/SRO/OK/DIRECT/RO23 AUDIT-016-3/006000 A2.11/4.0/4.4/H/3

Given the following conditions on Unit 1:

- Unit is responding to a LOCA using 1-OHP-4023-E-1, Loss of Reactor or Secondary Coolant.
- The Unit Supervisor is at Step 11, Initiate Evaluation Of Plant Status.

The following plant conditions now exist:

- RWST Level is 42% and lowering
- Containment pressure is 0.3 psig and stable
- Containment Recirc Sump Minimum Recirc Level Lights - NOT LIT
- East RHR Pump Compartment Sump Annunciator - LIT
- East RHR Pump Discharge pressure is 600 psig
- Aux Building area radiation monitors are in alarm

Which ONE of the following procedures should the Unit Supervisor transition into from 1-OHP-4023-E-1?

A.✓ 1-OHP-4023-ECA-1.2, LOCA Outside Containment

B. 1-OHP-4023-ECA-1.3, Sump Blockage Control Room Procedure

C. 1-OHP-4023-ECA-1.1, Loss of Emergency Coolant Recirculation

D. 1-OHP-4023-ES-1.3, Transfer To Cold Leg Recirculation

ANSWER: A

A - CORRECT. Based on RHR Pump Compartment Sump and Aux Building Rad alarms being in (along with the AEO report of the leak), transition to ECA-1.2, LOCA Outside Containment, would be required.

B - INCORRECT. Plausible since the Minimum Sump Recirc Lights are not lit, but the transition to ECA-1.3 is made only if the Red vortex alarm is lit.

C - INCORRECT. Transition to ECA-1.1 is only made in E-1 if one Train combination of a Recirc Sump Valve and its associated RHR pump is not verified available. Plausible if the student remembers a transition to this procedure exists from E-1 and some of the conditions exist to support that transition (however not all of the conditions exist).

D - INCORRECT. Transition to ES-1.3 is made per the foldout page when RWST level is less than 30%. Plausible as ES 1.3 is one of the highest level EOP procedures and the students know that this procedure will address 2 of the other 3 distractors by transitions from ES 1.3.

LESSON PLAN/OBJ: RO-C-EOP09/#40, 42
REFERENCE: 1-OHP-4023-E-1

KA - 006000 A2.11
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:
Rupture of ECCS header
RO - 4.0 SRO - 4.4
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Requires the SRO to predict that a LOCA outside Containment exists and that transition to ECA-1.2, LOCA Outside Containment, procedure is required.

Original Question # - RO23 AUDIT-016-3
Original Question KA - WE 04 EK3.2

88. 088 004/SRO/OK/MODIFIED/RO23 AUDIT-068-4/008000 A2.07/2.5/2.8/H/3

Given the following conditions on Unit 1:

- The Unit Supervisor entered 1-OHP-4022-016-001, Malfunction Of The CCW System, 5 minutes ago.

<u>The following plant conditions now exist:</u>

- Reactor Power is 8%
- All RCP Motor Bearing Lube Oil CCW flow low annunciators are lit.
- All flows and temperatures are STABLE as follows:

	RCP #11	**RCP #12**	**RCP #13**	**RCP #14**
o Upper Motor Bearing temps	189ºF	190ºF	205ºF	191ºF
o Motor Bearing Lube Oil CCW flow	4 gpm	4 gpm	3 gpm	3 gpm

Which ONE of the following describes the sequence of actions the Unit Supervisor will provide to the panel operators?

A. 1) Perform a rapid shutdown per 1-OHP-4021-001-006, Rapid Power Reduction
2) Trip #13 RCP immediately after the reactor is shutdown.
3) Close NRV-163, Loop 13 PZR Spray Control

B.✓ 1) Trip the reactor
2) Go to 1-OHP-4023-E-0, Reactor Trip or Safety Injection
3) Trip #13 RCP after Reactor Trip is verified
4) Close NRV-163, Loop 13 PZR Spray Control

C. 1) Trip the reactor
2) Go to 1-OHP-4023-E-0, Reactor Trip or Safety Injection
3) Trip all RCPs after Reactor Trip is verified
4) Close NRV-163 and NRV-164, Loop 13 and 14 PZR Spray Control

D. 1) Perform a rapid shutdown per 1-OHP-4021-001-006, Rapid Power Reduction
2) Trip all RCPs after the reactor is shutdown.
3) Close NRV-163 and NRV-164, Loop 13 and 14 PZR Spray Control

ANSWER: B

A - INCORRECT. Reactor and affected RCP must be tripped. Rapid shutdown is not appropriate. Plausible as the rapid down power will not take long from 8% power and the temperature of the RCP is stable. Student must know this temperature is above the trip setpoint.

B - CORRECT. 1-OHP-4022-016-001, Caution prior to Step 1 states that the affected RCP must be removed from service by performing Step 17 if motor bearing temperatures exceed 200ºF. Step 17 and associated note states that the order is to Trip the Reactor, go to E-0, verify reactor trip, then trip the AFFECTED RCP.

C- INCORRECT. Loss of CCW requires trip of Reactor and ALL RCPs, but CCW Malfunction only requires tripping the AFFECTED RCP (RCP #13 with the temperature >200ºF). Plausible as all actions are correct except stopping all RCP's. This is incorrect and will complicate the mitigation strategy of the EOPS but is a fairly routine activity in the simulator for the students to stop all RCP's vs. just stopping one RCP.

D - INCORRECT. Reactor and affected RCP must be tripped. Rapid shutdown is not appropriate. Combined the plausibility of answers A and C.

LESSON PLAN/OBJ: RO-C-AOP-D14/#RO-C-AOP0140412-E3,
RO-C-AOP-D8/#RO-C-AOP0420412-E2

REFERENCES: 1-OHP-4022-016-001, Malfunction Of The CCW System First Caution, 1-OHP-4021-002-001, RCP Malfunction, Step 1

KA - 008000 A2.07
Component Cooling Water System (CCWS)
Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Consequences of high or low CCW flow rate and temperature; the flow rate at which the CCW standby pump will start
RO - 2.5 SRO - 2.8
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Requires the ability to predict the impact of a loss of CCW flow will have on the RCPs and to direct the actions required based on the condition of the RCP following the loss of CCW flow.

Original Question # - RO23 Audit-068-4
Original Question KA - APE 026 G2.4.47

89. 089 003/SRO/OK/DIRECT/RO25 AUDIT-82/034000 K1.01/2.5/3.2/F/3

Given the following conditions:

- The refueling crew has just set a new fuel assembly next to an irradiated fuel assembly in the Unit 1 core.
- The manipulator crane operator observes the refueling cavity level lowering rapidly.
- The SRO-CA and Control Room begin implementing 1-OHP-4022-002-006, Loss of Refueling Water Level during Refueling Operations.

<u>The following conditions exist Twenty Minutes Later:</u>

- The Transfer Tube Gate Valve has been closed
- The Weir Gate could NOT be closed
- RCS level is 614' and lowering
- RWST to RHR makeup is in progress

Based on these conditions, which ONE of the following describes the correct actions for the Control Room SRO?

A. Go to 12-OHP-4022-018-001, Loss of SFP Cooling.

B.✓ Go to 1-OHP-4022-017-001, Loss of RHR Cooling

C. Direct the SRO-CA to verify integrity of the Refueling Cavity Seal

D. Direct the SRO-CA to check for misalignment of the Refueling Cavity Drains

ANSWER: B

A - INCORRECT. Since the Transfer Tube is Isolated, The Spent Fuel Pool has been isolated even if the Weir gate is not isolated. Plausible as this is a transition from the procedure in use.

B - CORRECT. If RCS Level can NOT be maintained >614' then an RCS leak is in Progress and 1-OHP-4022-017-001 Loss of RHR Cooling Must be Initiated.

C - INCORRECT. Even though the Refueling Cavity Drains/Piping are below the 621' elevation, the physical construction for the Refueling cavity will not allow an inadvertent draining of the Refueling Cavity to lower level below the Reactor Vessel Flange (~621' elevation). Plausible as this is a likely source of leakage.

D - INCORRECT. The Seal is > 620" elevation so even if it were failed the level drop would have stopped at 620'. Plausible as this is a likely source of leakage

LESSON PLAN/OBJ: RO-C-AOP-D9/#RO-C-AOP0130412-E3
REFERENCE: RO-C-AOP-D9; 1-OHP-4022-002-006

MODIFIED: Remove distractor relating to nozzle dams and replaced with distractor pertaining to Refueling cavity Drains.

KA - 034000 K1.01
Fuel Handling Equipment System (FHES)
Knowledge of the physical connections and/or cause-effect relationships between the Fuel Handling System and the following systems:
RCS
RO - 2.5 SRO - 3.2
CFR - 41.2 to 41.9 / 45.7 to 45.8

KA Justification - Requires the knowledge of the interconnections between the RCS water inventory and the Transfer Canal water inventory. Based on these interrelationships, question requires the SRO to determine appropriate procedure for a leak that affects both water inventories.

Original Question # - RO25 AUDIT-82
Original Question KA - 000036 2.4.4

90. 090 003/SRO/OK/DIRECT/ITS FINAL 3.7-21/039000 2.2.37/3.6/4.6/H/3

Given the following conditions on Unit 2:

- Unit is in MODE 4 and preparing to startup following a refueling outage.
- During the outage, all 5 Main Steam Safety Valves (MSSVs) associated with SG21 were removed and inspected.
- The maintenance was satisfactory and the MSSVs will function if required.
- The Inservice Testing of the MSSV setpoints has NOT been performed.
- A risk assessment has NOT been performed.

Without reliance on SR 3.0.3, which ONE of the following describes if the reactor startup can proceed to MODE 3 with the MSSVs in this condition?

Reference Provided

A. No. The TS ACTIONS must be immediately entered and all portions of the post testing must be completed before entering MODE 3.

B. No. An alternate method of setpoint verification must be used and MSSV OPERABILITY must be demonstrated before entering MODE 3.

C. Yes. However, when the unit reaches MODE 2, the test must be at least started within 24 hours after entering MODE 2.

D.✓ Yes. However, when test conditions can be established, the test must be completed prior to MODE 2.

ANSWER: D

A - INCORRECT. Entry into Mode 3 is permitted by the Note prior to SR 3.7.1.1. Plausible as the LCO applicability is Modes 1-3.

B - INCORRECT. Entry into Mode 3 is permitted by the Note prior to SR 3.7.1.1. Plausible as the LCO applicability is Modes 1-3 and distractor provides a logical method of compliance.

C - INCORRECT. Surveillance requirement must be complete PRIOR to entry into Mode 2. Plausible as this applies the missed Surveillance requirement to the distractor.

D - CORRECT. SR 3.7.1.1 NOTE states that the Surveillance Requirements are only required to be performed in MODES 1 and 2. Per Example 1.4-5, this note allows entry into Mode 3 to perform the surveillance.

LESSON PLAN/OBJ: RO-C-05103/#RO-C-05103-E11
REFERENCE: TS SR 3.0.1 and Note to SR 3.7.1.1; TS Example 1.4-5

Reference Provided: Unit 2 TS Section 1.4, Frequency, Section SR 3.0, Surveillance Requirements, and TS 3.7.1, Main Steam Safety Valves

KA - 039000 2.2.37
Main and Reheat Steam System (MRSS)
Equipment Control
Ability to determine operability and/or availability of safety related equipment.
RO - 3.6 SRO - 4.6
CFR - 41.7 / 43.5 / 45.12

KA Justification - Requires SRO to apply TS for MSSVs and determine operability and restrictions based on the status of the valves.

Original Question # - ITS FINAL 3.7-21
Original Question KA - 2.1.12

91. 091 007/SRO/OK/NEW/NEW/045000 2.4.35/3.8/4.0/F/3

Given the following conditions on Unit 2:

- A Main Generator fire has been confirmed by an AEO who reported that flames are coming out of the Unit 2 Main Generator Shaft.

After directing a Reactor/Turbine Trip and entering 2-OHP-4023-E-0, Reactor Trip Response, which ONE of the following would you direct the AEO to perform first?

Dispatch an AEO to:

A. Shut down the Main Generator Seal Oil System per 2-OHP-4021-080-002 Operation Of Shaft Seal Oil System

B.✓ Depressurize and Purge the Main Generator per 2-OHP-4022-053-002 Emergency Degassing Of The Electrical Generator

C. Start All Fire Pumps per 12-OHP-4021-066-001, Fire Protection System (Water) Operation.

D. Shut down the Generator Condition Monitor at 2-GCM-AARP-11, Generator Condition Monitor Auto Alarm Remote Panel

ANSWER: B

A - INCORRECT.	Eliminate fuel for fire (H2) first. Plausible as this removes a source of ignition from the fire area but this would allow H2 to escape into the fire area.
B - CORRECT.	The major concern is the fire. The source of the fire is the H2 gas from the generator. The first action must be to eliminate the source of the H2 by degassing the main generator.
C - INCORRECT.	Eliminate fuel for fire (H2) first. Plausible as this would aid in putting the fire out but all pumps are not required
D - INCORRECT.	Eliminate fuel for fire (H2) first. Plausible as this is a required action when degassing the generator.

LESSON PLAN/OBJ: RO-C-AOP-D14/RO-C-AOP0510412-E2
REFERENCE: 2-OHP-4022-053-002, Emergency Degassing Of The Electrical Generator

KA - 045000 2.4.35
Main Turbine Generator (MT/G) System
Emergency Procedures/Plan
Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.
RO - 3.8 SRO - 4.0
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Requires the knowledge of emergency task (emergency degas of main generator) associated with the Main Turbine Generator (MT/G) System.

Original Question # - New
Original Question KA - New

92. 092 003/SRO/OK/DIRECT/ITS FINAL 3.7-3/061000 A2.04/3.4/3.8/H/3

Given the following conditions on Unit 2:

- A plant heatup is in progress following a forced shutdown
- Reactor coolant average temperature at 450°F when the following Auxiliary Feedwater (AFW) trains become inoperable:
  - 0100 on July 7, TDAFW train is declared inoperable due to 2-MCM-221 steam supply valve being inoperable.
  - 1830 on July 8, 2E AFW train is declared inoperable.
  - 1900 on July 8, 2-MCM-221 steam supply valve is restored to OPERABLE status.

Including any extensions permitted by TS, and without re-entering a Technical Specification condition requiring a plant shutdown, the 2E AFW train must be restored to OPERABLE status by __________________.

Reference Provided

A. 0100 on July 10.

B. 0100 on July 17.

C.✓ 1830 on July 11.

D. 1830 on July 12.

ANSWER: C

A - INCORRECT. This is 3 days (72 hours) from the first event. The 72 hour clock for the 2E AFW pump started at the time the 2E pump was declared INOPERABLE.

B - INCORRECT. This is applying the TDAFP 10 days from discovery of failure to meet LCO.

C - CORRECT. 2E Pump 72 hour clock started when the 2E pump became INOPERABLE. The clock does not get reset with the OPERABILITY of the TDAFP.

D - INCORRECT. This applies the 72 hour clock for the 2E pump plus a 24 hour extension per TS Section 1.3, Completion Times.

LESSON PLAN/OBJ: RO-C-05600/#13, RO-C-TS01/RO-C-TS01-E11
REFERENCE: TS 3.7.5, Required Action B.1; TS Example 1.3-3

Reference Provided: Unit 2 TS 3.7.5 Auxiliary Feedwater System, TS Section 1.3, Completion Times

KA - 061000 A2.04
Auxiliary / Emergency Feedwater (AFW) System
Ability to (a) predict the impacts of the following malfunctions or operations on the AFW System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
pump failure or improper operation
RO - 3.4 SRO - 3.8
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Requires TS knowledge and ability to determine operability actions times for AFW pumps failures and to determine the required time per TS and procedures that an AFW pump must be made OPERABLE.

Original Question # - ITS FINAL 3.7-3
Original Question KA - 2.1.12

93. 093 005/SRO/OK/DIRECT/RO24 AUDIT-096-8/011000 A2.11/3.4/3.6/H/3

Given the following conditions on Unit 1:

- The plant is operating at 6% power preparing for Turbine roll.
- NLP-151, PZR Level Channel 1 failed 4 hours ago. The bistables have been tripped and all actions are complete as per 1-OHP-4022-013-010, Pressurizer Level Instrument Malfunction.
- PZR level is currently 40% on the remaining PZR Level channels.

Which ONE of the following describes the effects on the plant if NLP-153, PZR Level Channel 3 fails low and the affect on Unit Supervisor's decision to trip bistables for the Channel 3 failure?

Note: Assume NO operator actions.

Reference Provided

A.✓ Letdown will Isolate and heaters will de-energize.

Bistables may be tripped without causing a reactor trip. Power must remain less than 10%.

B. Letdown will remain in service and heaters will de-energize.

Bistables should NOT be tripped since a reactor trip will be generated. Power must remain less than 10%.

C. Letdown will remain in service and heaters will de-energize.

Bistables may be tripped without causing a reactor trip. Power must be reduced to less than 5%.

D. Letdown will Isolate and heaters will de-energize.

Bistables should NOT be tripped since a reactor trip will be generated. Power must be reduced to less than 5%.

ANSWER: A

A - CORRECT. With Channel 1 NLP-151 in the tripped condition the High level Rx Trip signal will be made up for 1 channel (1/2 coincidence on remaining channels). The level control selector switch for the pressurizer is in the 2/3 position with channel 3 NLP-153 as the controlling channel. When it fails low letdown will isolate and the heaters will de-energize. When the bistables are tripped, a reactor trip signal will be generated but it is blocked by P-7 (Reactor and Turbine power both below 10%). Plant startup can NOT continue. Power must be maintained below 10% (P-7).

B - INCORRECT. Letdown will isolate. Reactor will not trip (See Answer A). Plausible as student could be focused on Technical Specification impacts and not consider operational impacts causing Letdown isolation.

C - INCORRECT. Letdown will isolate. Power does not need to be reduced. Power just must remain less than P-7 (10%). Plausible as student could confuse Mode change requirements with Tech Spec compliance.

D - INCORRECT. Reactor will not trip (See Answer A). Power does not need to be reduced. Power just must remain less than P-7 (10%). Plausible as student could believe a trip signal could be generated with the additional bistable actuation.

LESSON PLAN/OBJ: RO-C-00202/#RO-C-00202-E17; RO-C-TS01/RO-C-TS01-E11
REFERENCE: SOD-00202-003, Pressurizer Level Control
TS 3.3.1 actions D & N. Table 3.3.1-1 Function 9

Reference Provided: Unit 1 TS Section 3.0 Limiting Condition for Operation Applicability and TS 3.3.1 Reactor Trip System Instrumentation

KA - 011000 A2.11
Pressurizer Level Control System (PZR LCS)
Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of
Failure of the PZR level instrument - low
RO - 3.4 SRO - 3.6
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Requires the ability to predict the impacts of multiple PZR Level channel failures (including a low failure), and to use the AOPs to determine the appropriate actions.

Original Question # - INPO - DIRECT 5314, NRC02-045-3 (SRO41), RO24 Audit-096-8
Original Question KA - 000028 - G2.2.22

94. 094 003/SRO/OK/DIRECT/RO25 AUDIT-78/194001 2.1.23/4.3/4.4/H/4

Given the following conditions on Unit 1:

- Unit is in Mode 6
- Refueling cavity is filled to 24.2 feet above flange
- Core reload is in progress
- RCS temperature is 82°F
- 1W RHR is OOS due to an Oil Leak
- 1E RHR train is in operation

A leak has been reported on the 1E RHR pump mechanical seal heat exchanger. To repair the leak, the RHR pump must be stopped. Maintenance estimates it will take 40 minutes to complete repairs.

1. How does this affect the ability to continue core reload?
2. What is the basis for having one RHR loop in operation in this condition?

A.✔ 1. Core reload must be stopped.
2. Provides for adequate RCS mixing and control of reactor coolant temperature.

B. 1. Core reload must be stopped.
2. Ensures that a core Keff of less than or equal to 0.95 is maintained during fuel handling operations.

C. 1. Core reload may continue provided no operations are permitted that would dilute the refueling cavity boron concentration.
2. Provide for adequate RCS mixing and control of reactor coolant temperature.

D. 1. Core reload may continue provided no operations are permitted that would dilute the refueling cavity boron concentration.
2. Ensures that a core Keff of less than or equal to 0.95 is maintained during fuel handling operations.

ANSWER: A

A - CORRECT. Because both pumps are inoperable, core loading must stop. If the pump were not inoperable, core loading is allowed to continue without RHR for up to one hour provided no change in Boron Concentration. This allows loading near the edges of the core where flow may interfere with setting fuel assemblies. The indication and control of temperature is one of the bases for this LCO. Maintaining core Keff </=0.95 is the bases of LCO 3.9.1 which is boron concentration during refueling operations. The confusion is that the RHR bases is for MIXING of the borated water to prevent potential criticality.

B - INCORRECT. The bases is for RCS Temperature control. Plausible as the first portion of the answer is correct and the second portion is the basis for maintaining the Mode.

C - INCORRECT. Core Reload Must be Stopped. Plausible as this combines some of the actions in the applicable spec but the actions are not in line with the required work activities.

D - INCORRECT. Core Reload Must be Stopped & The bases is for RCS Temperature control. Plausible as this combines portions of answers B and C.

LESSON PLAN/OBJ: RO-C-01700/#13 & 15
REFERENCE: Tech Specs & Bases 3.9.4

KA - 194001 2.1.23
Generic
Conduct of Operations
Ability to perform specific system and integrated plant procedures during all modes of plant operation.
RO - 4.3 SRO - 4.4
CFR - 41.10 / 43.5 / 45.2 / 45.6

KA Justification - Requires the ability to determine if refueling can continue based on equipment availability (Integrated Plant Operation) and the basis for limiting operations.

Original Question # - CATAWBA2005, RO25 AUDIT-78
Original Question KA - 000025 2.2.25

95. 095 095/SRO/OK/DIRECT/KEWAUNEE-962002/194001 2.1.43/4.1/4.3/H/3

Given the following conditions on Unit 2:

- A normal plant startup and power escalation to 100% was initiated following a refueling outage.
- The reactor achieved 100% rated power (3468 MWth) approximately 4 hours ago.
- MTI has informed the Unit Supervisor the Blowdown flow instrument for SG21 is reading approximately 100 gpm higher than actual blowdown flow.

The Unit Supervisor should:

A. direct the control room operator to raise power slightly since actual thermal power is less than PPC calculated thermal power of 3468 MWth.

B.✔ direct an immediate power reduction to ensure the 1 hour average does not exceed 3468 MWth.

C. order that no reactor power adjustments be made for the next 1 hour so an accurate 1-hour power average is obtained.

D. order that no reactor power adjustments be made for the next 4 hours and then make adjustments to power as required.

ANSWER: B

A - INCORRECT. The blowdown flow error will result in PPC derived thermal power to be nonconservatively low. This may result in actual thermal power already being greater than 3468 MWth. A power rise would only make matters worse.

B - CORRECT. The blowdown flow error will result in PPC derived thermal power to be nonconservatively low. This may result in actual thermal power already being greater than 3468 MWth. Power must be reduced to ensure plant is operating that less than 3468 MWth actual power.

C - INCORRECT. Holding power stable for 1 hour will only keep the plant operating with actual power greater than 3468 MWth for a longer period of time. Power must be reduced to less than 3468 MWth.

D - INCORRECT. Holding power stable for 4 hours will only keep the plant operating with actual power greater than 3468 MWth for a longer period of time. Power must be reduced to less than 3468 MWth.

LESSON PLAN/OBJ: RO-C-ADM02/ #24
REFERENCE: 2-OHP-4021-011-001, At-Power Operation, Including Load Swings

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

KA Justification - Requires ability to determine the affect on reactivity form an erroneous calculation on secondary flow (SG Blowdown miscalculated) and (SRO) determine the required actions based on the reactivity anomaly.

Original Question # - KEWAUNEE-962002, RO26 AUDIT-DRAFT-95
Original Question KA - 194001 2.1.7

96. 096 005/SRO/OK/DIRECT/CM-1140-31956/194001 2.2.40/3.4/4.7/H/3

Given the following conditions in Unit 2:

- The unit was stable at 95% power with Rod Control in AUTO
- Control Bank D is at 220 steps with AFD at -6.5%
- A HIGH failure of Power Range channel N41 occurred
- The reactor operator responded by placing Rod control in MANUAL 15 seconds after the event.

Which ONE of the following may require prompt crew actions to ensure continued compliance with Technical Specifications?

Reference Provided

A.✓ Axial Flux Difference (AFD)

B. Quadrant Power Tilt Ratio (QPTR)

C. Rod Insertion Limits (RIL)

D. Shutdown Margin (SDM)

ANSWER: A

A - CORRECT. AFD is close to the -7.2 limit of TDB 2-Figure 13.1, for 95% power. Based on the The operator response time and the maximum rod speed of 72 steps/minute, the rods would travel approximately 18 steps into the core. Rod motion would place the AFD outside the target band which is has a 15 minute requirement to restore AFD within the band or be below 90% in the next 15 minutes. TS 3.2.3, Axial Flux Difference (AFD)

B - INCORRECT. Failure will cause rod motion. Rod motion is a symmetrical event which should not lead to a QPTR concern. Plausible as there are actions required per Technical Specifications for QPTR but they are not prompt (12 hours).

C - INCORRECT. Rod insertion Limit at 95% is approximately 180 steps on control bank D. With rods starting at 220 and only traveling 18 steps, the minimum rod height for this event is 202 steps on control bank D, which is above the RIL requirements of Technical Specifications. Plausible RIL are a concern with any rod insertion with the unit at high power.

D - INCORRECT. As long as rods are above the RIL then SDM should be maintained. See Answer C justification. Plausible as students confuse SDM with rod motion and students know the SDM is short time frame LCO.

LESSON PLAN/OBJ: RO-C-AOP-D7/#RO-C-AOP0300412-E2
REFERENCE: U2 COLR (Cycle 18) and TDB 2-Figure 13.1, TS 3.2.3, Axial Flux Difference (AFD)

Reference Provided: U2 COLR (Cycle 18) and TDB 2-Figure 13.1, Target Band and ARM

KA - 194001 2.2.40
Generic
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

KA Justification - Requires the ability to apply TS 3.2.3, Axial Flux Difference (AFD).

Original Question # - CM-1140-31956
Original Question KA - 2.1.11

97. 097 002/SRO/OK/MODIFIED/NRC EXAM 2008-97/194001 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	420,000	6.2
1630	420,000	6.2
1700	350,000	5.8
1730	280,000	5.3
1800	260,000	4.8
1830	120,000	4.6
1900	90,000	4.3
1930	90,000	4.0
2000	90,000	3.9

At 2000 hours, while performing Emergency Operating Procedures, a step is encountered which states "Check PZR level - GREATER THAN 20% [24% ADVERSE]".

Which ONE of the following describes the required Pressurizer level and why?

A. 20% because the Containment Radiation levels are no longer above the Adverse setpoint requirement.

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. 20% because the Containment Pressure is no longer above the Adverse setpoint requirement

ANSWER: B

A - INCORRECT. The integrated dose is (1,015,000 R) which is greater than 10^6 R, so adverse containment values must be used. Plausible as the instant values for dose do not require the use of Adverse numbers

B - 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 (1,015,000 R) which is greater than 10^6 R, so adverse containment values must be used.

C - INCORRECT. The current Dose Rate is <10^5 R/Hr. Plausible as the instant value for Containment pressure is close to the adverse setpoint

D - INCORRECT. Pressure is <5 psig. Plausible as the instant value for containment pressure is close to the adverse setpoint.

LESSON PLAN/OBJ: RO-C-EOP01/#8 & #9
REFERENCE: OHI-4023, Abnormal / Emergency Procedure User's Guide, Attachment 2, Step 6, RO-C-EOP01

Modified: Raised dose rates so that the integrated dose becomes >10^6 R changing the correct answer to B. Added one more hour of readings. Changed Distractor A & D.

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

KA Justification - Requires the ability to use fixed rad monitor (Containment High Range) to determine whether adverse containment conditions exist.

Original Question # - Cook NRC Exam 2004 118-2(SRO87), NRC EXAM 2008-97
Original Question KA - 194001 2.3.5

98. 098 003/SRO/OK/DIRECT/AOP1CAOP7.10-1/194001 2.3.14/3.4/3.8/F/3

Which ONE of the following responses correctly reflects the bases for Reactor Coolant Specific Activity in Technical Specifications?

A. The short-lived radioactive isotope fission products will have decayed prior to any fuel movement.

B. Limitations on specific activity in the RCS reduces corrosion product activation and subsequent RCS integrity challenge.

C. During a LOCA the dose will NOT exceed the 10CFR20 limits at the site boundary.

D.✓ During a steam generator tube rupture the release of activity through the atmospheric relief valves will NOT exceed 10CFR100 limits.

ANSWER: D

A - INCORRECT. See Answer D. Plausible as this is a concern during refueling evolutions for dose to employees moving fuel.

B - INCORRECT. See Answer D. Plausible as this identifies a challenge to corrosion affecting the metal in the RCS for long term RCS operation.

C - INCORRECT. See Answer D. Plausible for accident dose concerns but the CFR reference is in correct.

D - CORRECT. The maximum dose to the whole body and the thyroid that an individual at the site boundary can receive for 2 hours during an accident is specified in 10 CFR 100 (Ref. 1). The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.
The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

LESSON PLAN/OBJ: RO-C-00200/# RO-C-00200-E10,
RO-C-AOP-D8/#RO-C-0030500401-E3
REFERENCE: TS 3.4.16 Bases

KA - 194001 2.3.14
Generic
Radiation Control
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
RO - 3.4 SRO - 3.8
CFR - 41.12 / 43.4 / 45.10

KA Justification - Requires the Bases (SRO) knowledge for the TS limits on RCS Activity relating the radiological hazards of a SGTR and release to the atmosphere.

Original Question # - CM-1166-31980, AOP1CAOP7.10-1
Original Question KA - 2.1.12

99. 099 003/SRO/OK/DIRECT/NRC EXAM 2002-025-12/194001 2.4.4/4.5/4.7/H/3

Given the following conditions on Unit 2:

- A LOCA has occurred
- The STA is monitoring the Critical Safety Functions and notes the following indications:

o WR log power	0%			
o WR startup rate	Negative			
o Containment Pressure	13 psig			
o CETC's 5 highest	760°F			
o RVLIS Narrow Range	76%			
o Pressurizer Level	0%			
o RCS Pressure	480 psig			
o AFW Flow	300 x10^3 pph			
	SG21	**SG22**	**SG23**	**SG24**
o Narrow Range SG Levels	12%	15%	16%	12%

Given the conditions described above, to which ONE of the following procedures should the SRO transition?

A. 2-OHP-4023-FR-C.2, Response to Degraded Core Cooling

B. 2-OHP-4023-FR-I.2, Response to Low Pressurizer Level

C✓ 2-OHP-4023-FR-Z.1, Response to High Containment Pressure

D. 2-OHP-4023-FR-H.1, Response to Loss of Secondary Heat Sink

ANSWER: C

A - INCORRECT. 2-OHP-4023-FR-C-2 is identified by RCS Temp >752°F and RVLIS >46% but it is an ORANGE path so it has a lower priority.

B - INCORRECT. 2-OHP-4023-FR-I-2 is indicated by Pressurizer level <17% but it is a YELLOW path so it has a lower priority.

C - CORRECT. Containment Pressure of >12 psig is a RED path requiring 2-OHP-4023-FR-Z-1.

D - INCORRECT. 2-OHP-4023-FR-H-1 is not indicated. ALL SGs are <24% (adverse) but AFW flow is sufficient and so the only H series procedure would be a YELLOW path for 2-OHP-4023-FR-H-5.

LESSON PLAN/OBJ: RO-C-EOP01/#23
REFERENCE: OHI-4023, Abnormal /Emergency Procedure User's Guide, Attachment 5, Critical Safety Function Status Trees

KA - 194001 2.4.4
Generic
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

KA Justification - Question tests the ability of the SRO to evaluate the plant conditions and determine the required procedural transition.

Original Question # - NRC Exam 2002-025-12

100. 100 004/SRO/OK/DIRECT/RO24 AUDIT 085-11/194001 2.4.5/3.7/4.3/F/3

Given the following conditions on Unit 2:

- The unit has tripped and experienced a safety injection.
- While performing 2-OHP-4023-ES-1.2, Post LOCA Cooldown and Depressurization, an ORANGE path condition was noted for the Core Cooling Critical Safety Function.
- 2-OHP-4023-FR-C.2, Response to Degraded Core Cooling, was entered.

While performing steps of this procedure, the Shift Technical Advisor reports the following:

- RED path condition exists for Core Cooling Critical Safety Function.
- RED path condition exists for Containment Critical Safety Function.
- NO other abnormal conditions were noted.

Based on these plant conditions, which ONE of the following is the appropriate action for the Unit Supervisor to take?

A. Complete actions of 2-OHP-4023-FR-C.2, Response to Degraded Core Cooling, and then transition to 2-OHP-4023-FR-Z.1, Response to Containment High Pressure.

B. Complete actions of 2-OHP-4023-FR-C.2, Response to Degraded Core Cooling, and then transition to 2-OHP-4023-FR-C.1.

C. Stop performing 2-OHP-4023-FR-C.2, Response to Degraded Core Cooling, and immediately transition to 2-OHP-4023-FR-Z.1, Response to Containment High Pressure.

D✓ Stop performing 02-OHP-4023-FR-C.2, Response to Degraded Core Cooling, and immediately transition to 2-OHP-4023-FR-C.1, Response to Inadequate Core Cooling.

ANSWER: D

A - INCORRECT. FR-C-1 is higher Priority than both FR-C.2 and FR-Z.1.
B - INCORRECT. FR-C-1 is higher Priority than FR-C.2
C - INCORRECT. FR-C.2 is higher priority than FR-Z.1
D - CORRECT. Per OHI-4023, Rules of usage, when a Red path is encountered, immediately initiate the FRP. Because FR-C. 1 is a higher priority than FR-Z. 1, the US should proceed to FR-C. 1 vs. FR-Z. 1.

LESSON PLAN/OBJ: RO-C-EOP01/#22
REFERENCE: OHI-4023 Abnormal/Emergency Procedure User's Guide, Attachment 5 Section 5

KA - 194001 2.4.5
Generic
Emergency Procedures/Plan
Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.
RO - 3.7 SRO - 4.3
CFR - 41.10 / 43.5 / 45.13

KA Justification - Requires knowledge of the priority (organization) of the Functional Restoration Procedures (FRPs) and the ability to apply this knowledge to determine the appropriate procedure to implement.

Original Question # - RO24 AUDIT 085-11
Original Question KA - W/E03.EA2.1