1. 001 007/RO- DONE/OK/NEW/NEW/001000 K5.30/RO-C-GF06-E9/RO-C-GF06-E9-1/F/3

Over core life (BOC to EOC), the differential rod worth will become ...

A. more negative (worth more) due to a lower boron concentration and a shift in the radial flux.

B. less negative (worth less) due to a lower boron concentration and a shift in the radial flux.

C. more negative (worth more) due to increased neutron absorption in moderator.

D. less negative (worth less) due to increased resonance absorption of neutrons.

ANSWER: A

A. CORRECT	Over core life the boron concentration is lowered allowing less competition for neutrons. Additionally as the flux shifts to the outer portion of the core more rods are affected (exposed to a higher flux) also making rod worth more negative.
B. INCORRECT	Rod Worth becomes more negative. Plausible since the reasons are correct.
C. INCORRECT	Rod worth becomes more negative but not due to the adsorption in the moderator (this would have opposite effects).
D. INCORRECT	Rod Worth becomes more negative. Plausible since the reasoning is partially true, absorbing more neutrons will lead to a more negative rod worth.

OBJECTIVE: RO-C-GF06-E9
REFERENCE: RO-C-GF06 (PR05I) pages 25-28
KA - 001000 K5.30
Control Rod Drive System
Knowledge of the operational implications of the following concepts as they apply to the CRDS:
Effects of fuel burnout on reactivity in the core
RO - 2.9 SRO - 3.1
CFR - 41.5 / 45.7

KA Justification - Question tests operator knowledge of the direction and reason of differential rod reactivity worth over core life.

Original Question #- NEW NRC Exam 2012-001
Original Question KA - N/A

2. 002 005/RO/OK/DIRECT/RO25 AUDIT-6/003000 A4.01/RO-C-00300-E19/NRCAUDIT07-0977/H/3

Given the following initial conditions on Unit 2:
- QRV-251 CCP Discharge Flow Control Valve is in MANUAL.
- QRV-200 Charging Header Pressure Control Valve is in MANUAL.
- Pressurizer pressure is 2235 psig.
- Reactor Coolant Pump seal injection flow is 32 gpm.
- Charging line flow is 89 gpm.

If pressurizer pressure is raised to 2300 psig, which ONE of the following sets of system parameter changes is correct?

A. Charging line flow lowers and total seal injection flow lowers.

B. Charging line flow lowers and total seal injection flow remains the same.

C. Charging pump discharge header pressure remains the same and total seal injection flow rises.

D. Charging pump discharge header pressure rises and total seal injection flow remains the same.

ANSWER: A

A. CORRECT	Centrifugal pump laws require that the discharge header pressure raises and flow lowers as system pressure rises. Therefore charging line flow and total seal flow will lower while charging line discharge pressure rises.
B. INCORRECT	The RCP Seals will also see the pressure rise causing the seal flow to also lower.
C. INCORRECT	Centrifugal pump laws require that the discharge header pressure raises and flow lowers as system pressure rises
D. INCORRECT	Centrifugal pump laws require that the discharge header pressure raises and flow lowers as system pressure rises

OBJECTIVE: RO-C-00300-E19
REFERENCE: SOD-00300-001, Charging and Letdown System
KA - 003000 A4.01
Reactor Coolant Pump System (RCPS)
Ability to manually operate and/or monitor in the control room:
Seal injection
RO - 3.3 SRO - 3.2
CFR - 41.7 / 45.5 to 45.8

KA Justification - The question requires the candidate to describe to expected response of the RCP Seal Injection flow based on changing conditions.

Original Question # - RO25 AUDIT-6, CATAWBA2005
Original Question KA - 000022 AK1.02

3. 003 003/RO/OK/NEW/NEW/003000 K6.04/RO-C-00201-E6/RO-C-00201-E6-1/H/2

Unit 1 is in Mode 3 at normal operating temperature and pressure.
While performing trouble shooting activities, CCM-453 (CCW from RCP Thermal Barrier) closes.

Which ONE of the following describes the impact this failure will have on the Reactor Coolant Pump(s) (RCP)?

Thermal Barrier Component Cooling Water return flow has been lost from ...

A. only 1 RCP. RCP operation may continue.

B. all RCPs. RCP operation may continue.

C. only 1 RCP. The affected RCP must be stopped within 5 minutes due to a loss of cooling.

D. all RCPs. All RCPs must be stopped within 5 minutes due to a loss of cooling.

ANSWER: B

A. INCORRECT	The thermal barrier isolation valve is common to all RCPs. Plausible since NESW cooling has seperate valves.
B. CORRECT	The CCW thermal barrier isolation valve is common to all RCPs. RCP operation may continue provided seal injection is maintained.
C. INCORRECT	The CCW thermal barrier isolation valve is common to all RCPs. Plausible since NESW cooling has seperate valves and a loss of CCW BEARING cooling does require the RCPs to be stopped.
D. INCORRECT	The CCW thermal barrier isolation valve is common to all RCPs. Plausible since NESW cooling has seperate valves and a loss of CCW BEARING cooling does require the RCPs to be stopped.

OBJECTIVE: RO-C-00201-E6
REFERENCE: RO-C-00201, pgs 14-17

KA - 003000 K6.04
Reactor Coolant Pump System (RCPS)
Knowledge of the effect of a loss or malfunction of the following will have on the RCPS:
Containment isolation valves affecting RCP operation
RO - 2.8 SRO - 3.1
CFR - 41.7 / 45.5

KA Justification - This question requires knowledge of the impact on the RCP thermal barrier CCW return flow from a failure of a containment isolation valve and the associated impact on RCP operation.

Original Question # - NEW NRC EXAM 2012-003
Original Question KA - N/A

4. 004 003/RO/OK/NEW/NEW/004000 K4.12/RO-C-00300-E12/RO-C-00300-E12-1/H/3

Given the following plant conditions:

- Unit 2 is operating at 100%.
- 120 GPM letdown is in service.
- Volume Control Tank (VCT) blender controls are in automatic.
- VCT level transmitter QLC-452 fails to 100%.

Which ONE of the following describes the expected impact to the VCT?

With NO operator action, the VCT level will lower ...

A. slowly due to normal RCS losses. Auto VCT makeup will NOT start and eventually the VCT level will lower to the RWST switchover setpoint. With QLC-452 failed high, the coincidence for switchover will not be made up and level will continue lowering to 0%. At this time suction will be lost to the charging pumps.

B. due to Letdown diverting. Auto VCT makeup will NOT start and eventually the VCT level will lower to the RWST switchover setpoint. QLC-451 will acuate a switchover to RWST Suction on Lo-Lo VCT level. At this time VCT level will start to rise due to seal return.

C. due to Letdown diverting. Auto VCT makeup will attempt to control VCT level. Eventually VCT level will lower to the RWST switchover setpoint. With QLC-452 failed high, the coincidence for switchover will not be made up and level will continue lowering to 0%. At this time suction will be lost to the charging pumps.

D. slowly due to normal RCS losses. Auto VCT makeup will attempt to control VCT. Eventually VCT level will lower to the RWST switchover setpoint. QLC-451 will acuate a switchover to RWST Suction on Lo-Lo VCT level. At this time VCT level will start to rise due to seal return and VCT Makeup.

ANSWER: C

A. INCORRECT	QLC-452 will cause full divert and does not impact makeup.
B. INCORRECT	QLC-452 does not impact makeup and switchover to the RWST requires both channels.
C. CORRECT	QLC-452 causes full divert, QLC-451 actuates makeup and switchover to the RWST requires both channels.
D. INCORRECT	QLC-452 causes full divert and switchover to the RWST requires both channels.

OBJECTIVE: RO-C-00300-E12
REFERENCE: SOD-00300-001

KA - 004000 K4.12
Chemical and Volume Control System (CVCS)
Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following:
Minimum level of VCT
RO - 3.1 SRO - 3.4
CFR - 41.7

KA Justification - This question requires knowledge of the required logic for lo-lo VCT level and the response of the system to a level channel failure.

Original Question # - NEW 2012-004-3
Original Question KA - N/A

5. 005 050/RO/OK/DIRECT/CM-0227/005000 K4.11/RO-C-00800-E7/CM-0227/F/3

It is desired to provide suction to the Safety Injection (SI) pumps from the Residual Heat Removal (RHR) Pumps during the Recirculation Phase. The RHR and Containment Spray (CTS) Pump suction valves from the Refueling Water Storage Tank (RWST) are already closed.

Which ONE of the following actions must be done to provide suction to the SI Pumps from RHR?

A. Open ICM-305 Recirc Sump to East RHR/CTS Pumps.
Close IMO-361 AND IMO-362 SI Pp Suction X-Tie to Charging Pumps Suction.
Open IMO-340 Charging Pump Suction from East RHR Hx.

B. Open ICM-306 Recirc Sump to West RHR/CTS Pumps.
Close IMO-361 AND IMO-362 SI Pp Suction X-Tie to Charging Pumps Suction.
Open IMO-350 SI Pp Suction from West RHR Hx.

C. Open ICM-305 Recirc Sump to East RHR/CTS Pumps.
Close IMO-262 OR IMO-263 SI Pps Recirc to RWST.
Open IMO-340 Charging Pp Suction from East RHR Hx.

D. Open ICM-306 Recirc Sump to West RHR/CTS pumps.
Close IMO-262 OR IMO-263 SI Pps Recirc to RWST.
Open IMO-350 SI Pp Suction from West RHR Hx.

ANSWER: D

A. INCORRECT The East RHR pump provides suction to the Charging pumps which may also share suction with the SI pumps but through the IMO-361 and 362 valves. IMO-262 OR IMO-263 SI Pps Recirc to RWST valves must be closed to allow opening IMO-340.

B. INCORRECT The West RHR pump provides suction to the SI pumps but the IMO-262 OR IMO-263 SI Pps Recirc to RWST valves must be closed to allow opening IMO-350.

C. INCORRECT The East RHR pump provides suction to the Charging pumps which may also share suction with the SI pumps but through the IMO-361 and 362 valves AND IMO-360. IMO-262 OR IMO-263 SI Pps Recirc to RWST valves must be closed to allow opening IMO-350.

D. CORRECT The West RHR pump provides suction to the SI pumps and closing either the IMO-262 OR IMO-263 SI Pps Recirc to RWST valves will allow opening IMO-350.

OBJECTIVE: RO-C-00800-E7
REFERENCE: SOD-00800-001

KA - 005000 K4.11
Residual Heat Removal System (RHRS)
Knowledge of RHRS design feature(s) and/or interlock(s) which provide for the following:
Lineup for low head recirculation mode (external and internal)
RO - 3.5 SRO - 3.9
CFR - 41.7

KA Justification - The question requires knowledge of the interlocks required to open the RHR to SI pumps to align for low head recirculation mode.

Original Question # - CM-0227
Original Question KA - SF2.006.K4.17, SF2.006.A4.07, SF3.006.A4.07

6. 006 051/RO/OK/DIRECT/CM-0089/006000 2.1.27/RO-C-00800-E1/CM-0089/F/3

Which ONE of the following would be a concern if the Core Exit Thermocouple (CETC) temperatures were approaching 2200°F?

A. A zircalloy-water reaction is accelerated at temperatures above 2200°F.

B. 2200°F is 500 degrees below the fuel cladding melting point.

C. Clad temperature at 2200°F correlates to a fuel centerline temperature at the fuel's melting point.

D. The cladding weakens due to a zirconium phase change (from a close-packed hexagonal structure to body-centered cubic) at temperatures above 2200°F.

ANSWER: A

A. CORRECT	Zircalloy-water reaction is accelerated at temperatures above 2200 °F.
B. INCORRECT	Zircalloy-water reaction is the concern - Cladding Melt temp is much higher.
C. INCORRECT	Fuel melt is a concern of closer to 5000 degrees centerline and is designed to remain < 4700 degrees at clad temps of 2200°F.
D. INCORRECT	The cladding weakens due to the zircalloy-water reaction as it oxidizes.

OBJECTIVE: RO-C-00800-E1
REFERENCE: RO-C-GF22 (PT09I) pgs 5-6
KA - 006000 2.1.27
Emergency Core Cooling System (ECCS)
Conduct of Operations
Knowledge of system purpose and/or function.
RO - 3.9 SRO - 4.0
CFR - 41.7

KA Justification - This question requires knowledge of the reasons for the 2200°F limit for the ECCS acceptance criteria (design function).

Original Question # - CM-0089
Original Question KA - P2.1.1

7. 007 004/RO/OK/NEW/NEW/006000 K2.01/RO-C-00800-E8/RO-C-00800-E8-1/F/2

Which ONE of the following describes the power supply for the Unit 1 **North** Safety Injection (SI) Pump?

A. Bus T11A

B. Bus T11B

C. Bus T11C

D. Bus T11D

ANSWER: D

A. INCORRECT. This is a Train B bus, it supplies the South SI pump.
B. INCORRECT. This is a Train B bus and it does NOT supply the pumps.
C. INCORRECT. This is a Train A bus but it does NOT supply the pumps.
D. CORRECT. The North SI and Train A pumps are supplied from T11D.

OBJECTIVE: RO-C-00800-E8
REFERENCE: SOD-08201-001

KA - 006000 K2.01
Emergency Core Cooling System (ECCS)
Knowledge of bus power supplies to the following:
ECCS pumps
RO - 3.6 SRO - 3.9
CFR - 41.7

KA Justification - Question tests knowledge of which bus supplies power to the North SI pump which is used as part of the ECCS.

Original Question # - New NRC Exam 2012-007
Original Question KA - N/A

8. 008 009/BOTH/OK/DIRECT/NRC EXAM 2007-29/007000 A2.01/RO-C-0020920412-E3/2007-0420/H/3

Given the following conditions on Unit 1:

- Unit was operating at 100% power when the turbine tripped.
- The reactor failed to automatically trip but was manually tripped.
- All other systems operated as expected.
- The Emergency procedures have been performed and the plant stabilized.
- A pressurizer Power Operated Relief Valve (PORV) was failed open and was isolated by closing the block valve.

Which ONE of the following represents the expected status of the Pressurizer Relief Tank (PRT) and the actions that must be taken to restore it to normal limits?

A. PRT Temperature = 100°F, Level = 15%, and Pressure = 14 psig
Open the Vent to depressurize and add water to cool the tank.

B. PRT Temperature = 140°F, Level = 84%, and Pressure = 12 psig
Reduce level and add water to cool & depressurize the tank.

C. PRT Temperature = 280°F, Level = 82%, and Pressure = 34 psig
Open the Vent to depressurize and add water to cool the tank.

D. PRT Temperature = 240°F, Level = 95%, and Pressure = 3 psig
Reduce level and add water to cool & depressurize the tank.

ANSWER: B

A - INCORRECT. The tank temperature and level are too low and the pressure is too high. The Vent will not open at this pressure.
B - CORRECT. PRT temperature is normally at Containment Temperature of ~100-110°F with level 80-84% and pressure of ~ 2-3 psig.
C - INCORRECT. Given this temperature and pressure the tank would be saturated. This is not expected to occur from a single discharge of the PORV. The vent will not open at this pressure.
D - INCORRECT. At this temperature pressure would need to be 10 psig. Level would not be expected to increase this much.

OBJECTIVE: RO-C-0020920412-E3
REFERENCE:
1-OHP-4022-002-009, Leaking Pressurizer Power Operated Relief Valve, pgs 1-2
1-OHP-4021-002-006, Pressurizer Relief Tank Operation, pgs 6-9

KA - 007000 A2.01
Pressurizer Relief Tank/Quench Tank System (PRTS)
Ability to (a) predict the impacts of the following malfunctions or operations on the PRTS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Stuck-open PORV or code safety
RO - 3.9 SRO - 4.2
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Tests knowledge of impact to the PRT due to a PORV discharge and the required actions to restore the PRT.

Original Question # - NRC Exam 2002-100-1, RO26 AUDIT-78, RO27 Audit -33, NRC EXAM 2007-29, 2007-0420, NRCAUDIT07-0213, NRCAUDIT07-0890, RO26-0013
Original Question KA - 007000 K4.01, SF5.007.A1.02, SF5.007.A2.02, SF5.007.A2.03

9. 009 005/RO/OK/DIRECT/NRC EXAM 2006-084-11/008000 K1.03/RO-C-01350-E4/NRCAUDIT07-0756/F/3

The following plant conditions exist on Unit 2:

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

Which ONE of the following describes the response of the CCW system and the required actions, if any, for this condition?

A. No automatic actions will occur since the West CCW pump is running. No Lineup changes are required, operation in this condition is allowed indefinitely.

B. No automatic actions will occur since the CRS-4401, West CCW HX Radiation Monitor is still functioning. The West CCW HX must be aligned so the CRV-412 Vent Valve will automatically close on a high radiation signal.

C. CRV-412, CCW Surge Tank Vent Valve, will automatically close. The West CCW HX must be aligned so the CRV-412 Vent Valve may be reopened.

D. CRV-412, CCW Surge Tank Vent Valve, will automatically close. No Lineup changes are required.

ANSWER: D

A - INCORRECT.	Plausible since the East CCW HX monitor may be associated with the East pump but either radiation monitor will cause the surge tank vent to isolate.
B - INCORRECT.	Plausible since the east monitor has failed and the west is still operational but an alarm or failure of either will cause the Surge tank vent to isolate.
C - INCORRECT.	Plausible since the Surge tank vent is isolated. Alignment to the West HX will not allow the vent to be reopened.
D - CORRECT.	2-CRV-412, CCW Surge Tank Vent Shutoff Valve closes on High Rad Level Alarm, Low Sample Flow, External Failure on CRS-4300/4400, Channel 4301 - East and/or Channel 4401-West. The automatic action is the closure of the Vent valve. The closure of the vent and monitor failure don't impact system operability, so operation may continue with this lineup.

OBJECTIVE: RO-C-01350-E4
REFERENCE: 12-OHP-4024-139 #29
KA - 008000 K1.03
Component Cooling Water System (CCWS)
Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems:
PRMS
RO - 2.8 SRO - 3.0
CFR - 41.2 to 41.9 / 45.7 to 45.8

KA Justification - This question tests the effects to the CCW system due to a PRMS failure.

Original Question #- NRCAUDIT07-0756, NRC Exam 2006-084-11 (Removed SRO Operability Calls), Audit RO23-074-3 (#068)
Original Question KA - SF8.008.A2.04, SYS 073 A1.01

10. 010 003/RO/OK/DIRECT/NRC EXAM 2002-081-2/011000 - K2.02/RO-C-00202-E10/NRCAUDIT07-0132/F/3

Which ONE of the following describes the design alignment which provides power to the pressurizer heaters during a loss of off-site power condition?

A. Group A1, A2, and A3 from the 2AB Emergency DG via bus T21B.

B. Group C1, C2, and C3 from the 2CD Emergency DG via bus T21D.

C. Group A1, A2, and A3 from the 21BD bus crosstie.

D. Group C1, C2, and C3 from the 21AC bus crosstie.

ANSWER: B -

A - INCORRECT	Group A heaters are tied to 2AB EDG via Bus T21A.
B - CORRECT	Group C heaters are supplied from the 21PHC transformer which is tied to the 2CD EDG via bus T21D.
C - INCORRECT	Group A heaters do not connect to 21BD bus.
D - INCORRECT	Group C heaters do not connect to 21AC bus.

OBJECTIVE: RO-C-00202-E10
REFERENCE: SOD-08201-001, Emergency Electrical Distribution
KA - 010000 K2.01
Pressurizer Pressure Control System (PZR PCS)
Knowledge of bus power supplies to the following:
PZR heaters
RO - 3.0 SRO - 3.4
CFR - 41.7

KA Justification - Question tests knowledge of the source of power to the pressurizer heaters.

Original Question #- NRCAUDIT07-0132, 01002C0209~1, NRC Exam 2002-081-2 (66/67)
Original Question KA - 010000K201

11. 011 008/RO/OK/DIRECT/RO-C-01100-E10-2/012000 K4.05/RO-C-01100-E10/RO-C-01100-E10-2/F/2

Given the following conditions in Unit 2:

- Reactor Power is above the Permissive P-8 setpoint.
- All systems are in AUTO.

Which ONE of the following sets of conditions describes the inputs and coincidence to the Reactor Protection system indicating that the Main Turbine Generator is tripped?

	Auto Stop Oil Press		Stop Valve Positions
A.	1/3 less than 62 psig	OR	3/4 less than 1% Open
B.	1/3 less than 800 psig	AND	4/4 less than 1% Open
C.	2/3 less than 62 psig	AND	3/4 less than 1% Open
D.	2/3 less than 800 psig	OR	4/4 less than 1% Open

ANSWER: D

A - INCORRECT. Coincidences are incorrect: ASO is 2/3; TSV is 4/4. Logic is an "OR" function. ASO pressure of 62 psig is previous plant value.
B - INCORRECT. ASO Coincidences are incorrect and logic is an "OR" function.
C - INCORRECT. Coincidences are incorrect: ASO is 2/3; TSV is 4/4. ASO pressure of 62 psig is previous plant value.
D - CORRECT. Coincidences are correct: ASO is 2/3; TSV is 4/4. Logic stated correctly as an "OR" function ASO Pressure is < 800 psig

OBJECTIVE: RO-C-01100-E10
REFERENCE: RQ-C-KNOW, pg 6
KA - 012000 K4.05
Reactor Protection System
Knowledge of RPS design feature(s) and/or interlock(s) which provide for the following:
Spurious trip protection
RO - 2.7 SRO - 2.9
CFR - 41.7

KA Justification - Questions tests knowledge of the turbine trip signal logic that is designed to prevent spurious trips.

Original Question #- 011
Original Question KA - SF7.012.K4.02

12. 012 003/RO/OK/DIRECT/30373 KEWAUNEE2006/013000 K5.01/RO-C-01100-E6/RO-C-01100-E6-1/F/2

Concerning the Engineered Safety Features (ESF) initiation instrumentation for Pressurizer pressure, there are _____(1)_____ channels that input to Safety Injection for _____(2)_____ independent safety trains of ESF.

	(1)	(2)
A.	3	2
B.	4	4
C.	4	2
D.	3	3

ANSWER: A

A - CORRECT. There are 4 channels Pressurizer pressure. The fourth channel inputs to Reactor Protection Trip (RPS) ONLY. There are 2 TRAINS for ESF actuation (A & B).

B - INCORRECT. ONLY 3/4 Pressurizer channels feed SI and there are ONLY 2 TRAINS for ESF actuation (A & B).

C - INCORRECT. ONLY 3/4 Pressurizer channels feed SI. There are 2 TRAINS for ESF actuation (A & B).

D - INCORRECT. 3/4 Pressurizer channels feed SI and there are ONLY 2 TRAINS for ESF actuation (A & B).

OBJECTIVE: RO-C-01100-E6, RO-C-01101-E8
REFERENCE: RO-C-01100, pg 47; RO-C-01101, pg 14

KA - 013000 K5.01
Engineered Safety Features Actuation System (ESFAS)
Knowledge of the operational implications of the following concepts as they apply to the ESFAS:
Definitions of safety train and ESF channel
RO - 2.8 SRO - 3.2
CFR - 41.5 / 45.7

KA Justification - Question tests the operational implications (channel/train arrangement) of safety trains and ESF channels.

Original Question #- INPO Bank 30373, KEWAUNEE-2-2-2006-13
Original Question KA - 013000 K5.01

13. 013 003/RO/OK/DIRECT/NRCAUDIT07-0633/015000 2.2.37/RO-C-01300-E6/NRCAUDIT07-0633/H/3

The following plant conditions exist:

- With Unit 1 operating at 100% power and Unit 2 is shutdown in Mode 5.
- A Loss of Instrument CRID III occurred on Unit 2.
- The Unit 2 operators have taken their actions to verify proper shutdown condition.

What effect does this have on the Unit 1's ability to monitor reactor power?

A. NI Neutron Flux Power Level recorder SG-12 is lost which requires alternate monitoring of power level.

B. NI Neutron Flux Power Level recorder NR-43 is lost which requires alternate monitoring of power level.

C. Gamma Metirc instrument N23 is inoperable which reduces the post accident monitoring capability to only one available Nuclear Instrument.

D. Gamma Metirc instrument N21 is inoperable which reduces the post accident monitoring capability to only one available Nuclear Instrument.

ANSWER: C

A. INCORRECT	NI Instrument Panel 1-NIS-III powers N-23 from the opposite unit.
B. INCORRECT	NI Instrument Panel 1-NIS-III powers N-23 from the opposite unit.
C. CORRECT	Gamma Metirc instrument N23 is inoperable since it is powered from the opposite unit.
D. INCORRECT	NI Instrument Panel 1-NIS-III powers N-23 from the opposite unit. N21 is powered from its own unit CRID III.

OBJECTIVE: RO-C-01300-E6
REFERENCE: SD-01300 Excore Nuclear Instrumentation System, pg 25
KA - 015000 2.2.37
Nuclear Instrumentation System
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 - Question tests ability to determine availability of instrumentation based on a loss of power.

Original Question # - NRCAUDIT07-0633
Original Question KA - 000057 AA2.08

14. 014 005/RO/OK/DIRECT/NRC EXAM 2007-089/022000 K3.01/RO-C-AOP0480412-E3/RO26-0168/F/3

Unit 2 Containment Chill Water is operating on the OPEN loop (NESW) configuration when a Non-Essential Service Water (NESW) rupture inside containment occurs.

The crew has entered 2-OHP-4022-020-001, NESW System Loss/Rupture.
Which ONE of the following describes the required action(s) and the reason(s) for this/these action(s)?

The Unit Supervisor should direct the crew to trip the Reactor and ...

A. stop all RCPs to minimize the risk of fire since RCP fire protection has been lost.

B. stop all RCPs to prevent pump damage since all RCP cooling has been lost.

C. stop three RCPs. A containment pressure relief is performed to minimize the risk of a safety injection actuation since containment cooling has been lost.

D. stop three RCPs. A containment pressure relief is performed to allow containment purge supply to be started since ice condenser cooling has been lost.

ANSWER: C

A. INCORRECT	RCP fire protection is lost but this is not the reason for stopping RCPs. One RCP is maintained operating to aid in a cooldown.
B. INCORRECT	RCP motor air coolers are lost, but all cooling is not lost. The primary function of the motor air coolers is to cool the hot exhaust air from the RCP to keep the environment cool and not the pump. One RCP is maintained operating to aid in a cooldown.
C. CORRECT	The reactor must be tripped if a loss of NESW to containment occurs. Three RCPs are removed from service to stop heat input to the containment atmosphere during a loss of containment cooling. The heat input would cause a rapid rise in containment pressure, resulting in an SI and CTS actuation based solely on a loss of containment cooling.
D. INCORRECT	Ice condenser cooling is provided by the Glycol cooling system which uses NESW to Cool its chillers. A rupture inside containment should not impact this cooling. The containment purge system is not used.

OBJECTIVE: RO-C-AOP0480412-E3
REFERENCE: RO-C-AOP-D11--ATT-1, ILT AOP Classroom Day 1, pgs 34-44

KA - 022000 K3.01
Containment Cooling System (CCS)
Knowledge of the effect that a loss or malfunction of the CCS will have on the following: Containment equipment subject to damage by high or low temperature, humidity, and pressure
RO - 2.9 SRO - 3.2
CFR - 41.7 / 45.6

KA Justification - Question tests knowledge of the impact the loss of CSS has on containment systems and equipment.

Original Question # - 2007-0456, RO26-0168, RO26 AUDIT-58, Cook NRC Exam 2007-089, COOK04-084
Original Question KA - 022000 A2.05

Added "Unit 2 Containment Chill Water is currently on OPEN loop (NESW) configuration" to stem (NESW backs up CCW to CNTMT).

15. 015 005/RO/OK/NEW/NEW/025000 K5.02/RO-C-01000-E2/RO-C-01000-E2-1/F/2

Which ONE of the following correctly describes the operation of the Ice Condenser Air Handling Unit Fans?

A. The Containment Cooling System is used to maintain Ice Condenser temperature within limits and the Ice Condenser Air Handling Unit Fans must be STARTED when the Distributed Ignition System (DIS) is placed in service.

B. The Ice Condenser Air Handling Unit Fans are run to maintain Ice Condenser temperature within limits but must be STOPPED when Distributed Ignition System (DIS) is placed in service.

C. The Ice Condenser Air Handling Unit Fans are run to maintain Ice Condenser temperature within limits but must be STOPPED on High Containment Pressure to allow steam flow up through the ice condenser.

D. The Containment Cooling System is used to maintain Ice Condenser Temperature within limits and the Ice Condenser Air Handling Unit Fans must be STARTED on High Containment Pressure to direct steam flow up through the ice condenser.

ANSWER: B

A. INCORRECT	The Containment Cooling System maintains containment temperature only and the Air handling Unit Fans are STOPPED when DIS is put in service.
B. CORRECT	The Ice Condenser Air Handling Unit Fans are run to maintain Ice Condenser temperature within limits but must be STOPPED when DIS is placed in service.
C. INCORRECT	The Ice Condenser Air Handling Unit Fans are run to maintain Ice Condenser temperature within limits but do not have to be stopped to allow steam flow through the ice condenser.
D. INCORRECT	The Containment Cooling System maintains containment temperature only and the steam flow through the ice condenser is directed via turning vanes.

OBJECTIVE: RO-C-01000-E2
REFERENCE: RO-C-01000, pg 22

KA - 025000 K5.02
Ice Condenser System
Knowledge of the operational implications of the following concepts as they apply to the Ice Condenser System:
Heat transfer
RO - 2.6 SRO - 2.8
CFR - 41.5 / 45.7

KA Justification - The question requires knowledge of the operations required to provide the normal heat removal (transfer) and the method of providing accident heat transfer.

Original Question # - NEW NRC EXAM 2012-015
Original Question KA - N/A

16. 016 007/RO/OK/DIRECT/NRC EXAM 2002-54/026000 A4.01/RO-C-00900-E12/NRCAUDIT07-1047/H/3

The following plant conditions exist:

- A Unit 2 LOCA is in progress.
- Containment Pressure is 8.5 psig.
- IMO-210/211/220/221 (CTS Pump Discharge Valves) are Open.
- IMO-202/204 (SAT Outlet Valves) are Open.
- IMO-212/222 (SAT Eductor Valves) are Open.
- East Containment Spray (CTS) Pump is Running.
- West Containment Spray (CTS) Pump is NOT Running.
- Both RHR Pumps are Running.
- All SG Stop Valves are Closed.
- Panel 205, Drop 5, CONTAINMENT SPRAY ACTUATED, alarm actuated.
- Panel 205, Drop 10, CONTAINMENT ISOLATION PHASE B, alarm actuated.

Which ONE of the following failures would result in the above listed conditions?

A. Failure of Train A, Containment Isolation Phase B relay to actuate.

B. Failure of Train B, Containment Isolation Phase B relay to actuate.

C. Failure of Train A, Containment Spray (CTS) relay to actuate.

D. Failure of Train B, Containment Spray (CTS) relay to actuate.

ANSWER: B

A. INCORRECT Failure of Train A, Containment Isolation Phase B relay to actuate would have prevented the East CTS pump from starting.

B. CORRECT The Train B CTS pump is the West CTS Pump. The CTS pumps are actuated from the respective train Phase B signal while the associated valves reposition based on the CTS signal.

C. INCORRECT Failure of Train A, Containment Spray (CTS) relay to actuate would have prevented the IMO-202, 210, 211, and 212 valves from opening.

D. INCORRECT Failure of Train B, Containment Spray (CTS) relay to actuate would have prevented the IMO-204, 220, 221, and 222 valves from opening.

OBJECTIVE: RO-C-00900-E12
REFERENCE: RO-C-00900, pgs 20-21, Slide 48

KA - 026000 A4.01
Containment Spray System (CSS)
Ability to manually operate and/or monitor in the control room:
CSS controls
RO - 4.5 SRO - 4.3
CFR - 41.7 / 45.5 to 45.8

KA Justification - This question requires the operator to monitor the indications and determine an improper alignment.

Original Question # - NRC Exam 2002-54, RO25 AUDIT-39
Original Question KA - SF5.026.A3.01

17. 017 007/RO/OK/DIRECT/RO23 AUDIT-042/026000 K3.02/RO-C-01700-E2/NRCAUDIT07-0403/F/4

Unit 1 has experienced a large break LOCA.
The following conditions exist:

- OHP-4023-E-0, Reactor Trip or Safety injection, is complete.
- OHP-4023-ES-1.3, Transfer to Cold Leg Recirc, is complete.
- Only 1 Train of Containment Spray was available so RHR Spray was aligned as follows:
  - ICM-311 (RHR Discharge Isolation) is CLOSED.
  - IMO-330 (RHR to Upper Containment Spray) is OPEN.
- The remainder of the system is aligned as expected for this condition.

Which ONE of the following best describes the effects on the RHR system if the East RHR pump trips?

A. RHR Flow is lost to Cold Legs 2 & 3
RHR flow is still delivered to Cold Legs 1 & 4 and RHR Spray.

B. RHR Flow is lost to the RHR Spray.
RHR flow is still delivered to core through all 4 Cold Legs

C. RHR Flow is lost to the core through all 4 Cold Legs
RHR flow is still delivered to RHR Spray.

D. RHR Flow is lost to the RHR Spray.
RHR flow is still delivered to Cold Legs 2 & 3.

ANSWER: D
ICM-311 isolates East RHR From Cold Legs 1 & 4 (SI can still inject into 1&4)
IMO-330 is East RHR to spray.

A. INCORRECT ICM-311 isolates East RHR From Cold Legs 1 & 4 and the West RHR will supply Cold Legs 2 & 3.
B. INCORRECT ICM-311 isolates East RHR From Cold Legs 1 & 4.
C. INCORRECT The West RHR will supply Cold Legs 2 & 3 but spray flow from the East RHR is lost.
D. CORRECT The West RHR will supply Cold Legs 2 & 3 but spray flow from the East RHR is lost.

OBJECTIVE: RO-C-01700-E2
REFERENCE: SOD-00800-002 ECCS - Recirculation Phase

KA - 026000 K3.02
Containment Spray System (CSS)
Knowledge of the effect that a loss or malfunction of the CSS will have on the following:
Recirculation spray system
RO - 4.2 SRO - 4.3
CFR - 41.7 / 45.6

KA Justification - This question Requires knowledge of the required RHR alignment to support recirculation and CSS (due to a CTS failure) and the plant status if a RHR pump trips.

Original Question # - RO23 AUDIT-042, NRCAUDIT07-0403
Original Question KA - APE.025.AK2.02

18. 018 004/RO/OK/DIRECT/NRC EXAM 2006-048-10/028000 K2.01/RO-C-00900-E9/NRCAUDIT07-0717/H/3

Unit 1 has experienced a LOCA and Loss of Offsite power.

The following conditions exist:

- Emergency Diesel Generator 1AB failed to start.
- Emergency Diesel Generator 1CD has started and loaded as designed.
- Power has been restored to the Reserve Aux Transformers (RAT).
- No buses have been energized from the RATs.

The Unit Supervisor directs you to verify or restore power so a hydrogen recombiner may be run.

Which ONE of the following actions is required to enable the associated Hydrogen Recombiner to be operated?

A. Verify that bus T11C has energized 600V Bus 11C and MCC-1-EZC-C for (Train A) Hydrogen Recombiner Number 2.

B. Verify that bus T11C has energized 600V Bus 11C and Close the 11AC crosstie to supply power to Bus 11A and MCC-1-EZC-A for (Train B) Hydrogen Recombiner Number 1.

C. Energize RCP Bus 1B from the RAT to supply power to 600V Bus 11BMC for (Train B) Hydrogen Recombiner Number 1.

D. Energize RCP Bus 1C from the RAT to supply power to 600V Bus 11CMC for (Train A) Hydrogen Recombiner Number 2.

ANSWER: A

A. CORRECT	Hydrogen Recombiner #1 is powered form MCC-1-EZC-B and Hydrogen Recombiner #2 is powered form MCC-1-EZC-C.
B. INCORRECT	Hydrogen Recombiner #1 is powered form MCC-1-EZC-B.
C. INCORRECT	Hydrogen Recombiner #1 is powered form MCC-1-EZC-B.
D. INCORRECT	Hydrogen Recombiner #2 is powered form MCC-1-EZC-C.

OBJECTIVE: RO-C-00900-E9
REFERENCE: RO-C-00900, Containment Spray and Hydrogen Recombiner System Description, pg. 14-15

KA - 028000 K2.01
Hydrogen Recombiner and Purge Control System (HRPS)
Knowledge of bus power supplies to the following:
Hydrogen recombiners
RO - 2.5 SRO - 2.8
CFR - 41.7

KA Justification - Question tests knowledge of the power supply to the Hydrogen Recombiner.

Original Question # - NRCAUDIT07-0717, NRC Exam 2006-048-10, INPO # 27678
Cook. 1-04/29/2004(RO#044/SRO#044)
Original Question KA - 028000K201

19. 019 002/RO/OK/MODIFIED/NRC EXAM 2008-72/029000 K1.01/RO-C-01350-E4/RO-C-01350-E4-13/H/3

Given the following plant conditions on Unit 1:

- Containment Purge System is operating in the PURGE (clean-up) MODE.
- A power supply spike caused a HIGH alarm on ERS-1305, Noble Gas Low Range Containment Monitor.

Which ONE of the following describes the response of the Containment Ventilation System to this alarm?

A. Containment ventilation isolation valves VCR-101 through VCR-106 close.
HV-CIPS-1 Containment Instrument Room Purge Supply Fan trips.

B. Containment ventilation isolation valves VCR-101 through VCR-106 close.
HV-CPS-1/2 Containment Purge Supply Fans 1 and 2 trip.
HV-CPX-1/2 Containment Purge Exhaust Fans 1 and 2 trip.

C. Containment ventilation isolation valves VCR-201 through VCR-206 close.
HV-CIPX-1 Containment Instrument Room Purge Exhaust Fan trips.

D. Containment ventilation isolation valves VCR-201 through VCR-206 close.
HV-CPS-1/2 Containment Purge Supply Fans 1 and 2 trip.
HV-CPX-1/2 Containment Purge Exhaust Fans 1 and 2 trip.

ANSWER: A

A. CORRECT While Operating in the Clean-UP mode the Radiation Monitor switches are unblocked allowing actuations. These are the automatic actions from the ERS-1305 monitor.

B. INCORRECT. The Purge supply and exhaust fans will NOT trip.

C. INCORRECT. The Outside containment isolation valves (200 series) will NOT close.

D. INCORRECT. ERS-1405 closes the Outside Containment Isolation valves and trips the listed fans.

OBJECTIVE: RO-C-02800-E9; RO-C-01350-E4
REFERENCE: 12-OHP-4024-139, Drop 3

KA - 029000 K1.01
Containment Purge System (CPS)
Knowledge of the physical connections and/or cause-effect relationships between the Containment Purge System and the following systems:
Gaseous radiation release monitors
RO - 3.4 SRO - 3.7
CFR - 41.2 to 41.9 / 45.7 to 45.8

KA Justification - Question requires candidate to know the high alarm auto actions of the gaseous radiation release monitor affecting Containment Purge.

Original Question # - MODIFIED FROM RO27 Audit -72, NRC EXAM 2008-72, 2008 NRC-0637, NRC EXAM 2004-100-3, AUDIT 02-BOTH31
Original Question KA - 194001 2.3.15

Modified by changing to ERS-1305 Gas Monitor vs. Area Monitor. This Changes the Correct Answer to A vs. D.

20. 020 002/RO/OK/DIRECT/RO27 AUDIT-61/033000 K4.03/RO-C-01800-E3/RO27AUDIT-61/F/3

Given the following conditions :

- The North Spent Fuel Pit (SFP) Pump is stopped with the discharge valve closed.
- The North Spent Fuel Pit Pump suction header ruptures.

Which ONE of the following describes how far the SFP level would drop before the operating SFP pump loses suction?

The level would drop approximately _________ below the normal water level.

A. 4 inches

B. 19 inches

C. 6 feet

D. 19 feet

ANSWER: B

A - INCORRECT. The SFP Skimmer pumps take a suction from approximately 4 inches below normal level.

B - CORRECT. The SFP is designed and located to minimize the probability and effects of pipe ruptures. The suction for the SFP Cooling System ~19 inches below the normal SFP water level elevation to prevent draining of the SFP in the event of a suction line break.

C - INCORRECT. Plausible, since the pump discharge lines terminate (626' 1½") no lower than 6’ above the top of the fuel assemblies and have anti-siphon holes located about 4" below normal level.

D - INCORRECT. Plausible, since the pump discharge lines terminate (626' 1½" ~ 19 feet below normal) no lower than 6’ above the top of the fuel assemblies and have anti-siphon holes located about 4" below normal level.

RO-C-01800/KO-3: List the design features, which prevent inadvertent draining of the spent fuel pit.

- There are no gravity drains.
- The suction lines (SFP pump) are located (643' 6") just 19½" below normal water level (645' 1½").
- The pump discharge lines terminate (626' 1½") no lower than 6’ above the top of the fuel assemblies and have anti-siphon holes located about 4" below normal level.
- The adjustable support for the skimmers limits the downward travel to 6" below normal level.

OBJECTIVE: RO-C-01800-E3
REFERENCE: SOD-01800-001

KA - 033000 K4.03
Spent Fuel Pool Cooling System (SFPCS)
Knowledge of Spent Fuel Pool Cooling System design feature(s) and/or interlock(s) which provide for the following:
Anti-siphon devices
RO - 2.6 SRO - 2.9
CFR - 41.7

KA Justification - Requires knowledge of the SFP design features which ensure that the level is maintained sufficient to cover fuel.

Original Question #- RO27 Audit-61, WattsBarMay2009-610
Original Question KA - 033000 K4.01

21. 021 006/RO/OK/DIRECT/NRC EXAM 2002-77-1/034000 K6.02/RO-C-02801B-E8/NRCAUDIT07-0235/F/3

As an irradiated fuel assembly is being lifted with the Spent Fuel Handling Crane, R-5, the Spent Fuel Pool Fuel Handling Building area radiation monitor, goes into HIGH alarm.

Which ONE of the following describes the automatic actions associated with this alarm?

A. Fuel Handling Area Supply Fans 12-HV-AFS-1, 2, 3, 4 - START
AFX Filter Bypass Dampers - OPEN
AFX Filter Outlet Dampers - OPEN

B. Spent Fuel Handling Crane upward motion is BLOCKED.

C. Fuel Handling Area Supply Fans 12-HV-AFS-1, 2, 3, 4 - TRIP
AFX Filter Bypass Dampers - Remain as is.
AFX Filter Outlet Dampers - Remain as is.

D. Control Room (CR) A/C Intake Dampers HV-ACRDA-1, 1A - CLOSE
CR Pressurization Outside Air Intake Dampers HV-ACRDA-2, 2A - OPEN
CR Pressurization Recirc Damper HV-ACRDA-3 - OPEN

ANSWER: C

A. INCORRECT	The Supply fans trip on a high alarm. Exhaust air is drawn through the filters.
B. INCORRECT	Spent Fuel Handling Crane movement is not blocked. Motion would be procedurally stopped.
C. CORRECT	On HIGH Alarm, R-5 will cause the Fuel Handling Supply fans to trip and the Charcoal Filter to align (Bypass closes, Outlet opens). However, when fuel movement is taking place the charcoal filter is required to be in service and bypass dampers closed.
D. INCORRECT	Control Ventilation does not automatically realign from Spent fuel monitor.

OBJECTIVE: RO-C-02801B-E8
REFERENCE: 1-OHP-4024-138, Annunciator #138 Response: Electro-Larm, Drop #5

KA - 034000 K6.02
Fuel Handling Equipment System (FHES)
Knowledge of the effect of a loss or malfunction of the following will have on the Fuel Handling System:
Radiation monitoring systems
RO - 2.6 SRO - 3.3
CFR - 41.7 / 45.7

KA Justification - Question tests the knowledge of the effect on fuel handling ventilation from a malfunction/high alarm on the radiation monitor.

Original Question # - INPO Modified 9239, NRC Exam 2002-77-1, NRCAUDIT07-0235
Original Question KA - 072000 - K3.02

22. 022 004/RO/OK/DIRECT/RO27AUDIT-44/039000 A3.02/RO-C-01100-E6/RO27AUDIT-44/H/3

Given the following conditions on Unit 1:

- A reactor heatup is in progress, per OHP-4021-001-001, Plant Heatup from Cold Shutdown to Hot Standby.
- RCS temperature is stable at 485°F.

Which ONE of the following will automatically close the Steam Generator Stop Valves?

A. Steam line pressure has lowered to 500 psig.

B. Manual Containment Spray (CTS) actuation.

C. Containment pressure 1.1 psig on 2/3 channels.

D. High Steam Flow of 1.7e+6 pph.

ANSWER: D

A - INCORRECT. Plausible, since the logic and coincidence for the actuation of the Low Steam Header Pressure SI are correct, but since the plant is less than P-12, this function is manually blocked and does not reset until greater than P-12.

B - INCORRECT. Plausible, since an AUTO CTS actuation will automatically close the SG MSSVs.

C - INCORRECT. Plausible, since a High-High containment pressure on 2/4 channels will initiate and AUTO CTS/Phase B and close the SG MSSVs.

D - CORRECT. The plant is less than P-12, so the high steam flow isolation from high steam flow is enabled.

OBJECTIVE: RO-C-01100-E6
REFERENCE: RO-C-01100 pg. 37-38 & 52, RQ-C-KNOW pg. 7

KA - 039000 A3.02
Main and Reheat Steam System (MRSS)
Ability to monitor automatic operation of the MRSS, including:
Isolation of the MRSS
RO - 3.1 SRO - 3.5
CFR - 41.7 / 45.5

KA Justification - Requires the ability to determine when a Main Steam Isolation (isolation of the MRSS) will occur.

Original Question # - RO27AUDIT-44, WattsBarMay2009-44
Original Question KA - 039 A3.02

23. 023 003/RO/OK/DIRECT/RO-C-05501-E5-1/039000 A4.03/RO-C-05501-E5/RO-C-05501-E5-1/H/3

With Unit 1 operating at 100% power, FPC-250A, Channel A Feed Pump Discharge Header Pressure, begins to drift LOW in comparison with FPC-250B, Channel B Feed Pump Discharge Header Pressure.

Which ONE of the following describes the response of the Main Feed Pumps assuming both feed pumps are operating in AUTOMATIC Differential Pressure (DP) control?

A. No change in Feed Pump DP since the program uses the highest of the two channels to control Feed Pump DP.

B. No change in Feed Pump DP since FPC-250A is automatically disabled upon a deviation.

C. Actual Feed Pump DP rises slowly since the AVERAGE of FPC-250A and FPC-250B is used for indicated Feed Pump DP.

D. Actual Feed Pump DP rises slowly since the LOWER of FPC-250A and FPC-250B is used for indicated Feed Pump DP.

ANSWER: D

A. INCORRECT	Lowest reading of FPC is used. This would speed up MFP and raise delta-p.
B. INCORRECT	On a slow failure, neither channel is disabled until one exceeds high or low limit.
C. INCORRECT	Lowest reading of FPC is used not the average.
D. CORRECT	Lowest reading of FPC is used to be conservative on delta-p input when compared to highest UPC-101 reading. This would speed up MFP and raise delta-p.

OBJECTIVE: RO-C-05501-E5
REFERENCE: 4024-MFP-DCS Drop C-11 &14, TS3000- DCS BODD Page 4-6

KA - 039000 A4.03
Main and Reheat Steam System (MRSS)
Ability to manually operate and/or monitor in the control room:
MFW pump turbines
RO - 2.8 SRO - 2.8
CFR - 41.7 / 45.5 to 45.8

KA Justification - Question tests ability to monitor and predict changes associated with the MFW Pump Turbine Speed.

Original Question #- RO-C-05501-E5-1
Original Question KA - SF4.059.A3.07

24. 024 003/RO/OK/MODIFIED/NRC EXAM 2008-093/056000 A2.04/RO-C-05400-E8/2008NRC-0419A/H/3

Given the following plant conditions on Unit 2:
- The unit is at 50% power.
- East and West Main Feed Pumps (MFPs) are running.
- North and South Condensate Booster Pumps (CBPs) are running.

Then the following alarms are received:
- Ann. 216, Drop 72, CNDST BOOSTER PUMP MOTOR OVERLOAD TRIP.
- Ann. 216, Drop 73, CNDST BOOSTER PUMP DISCH PRESSURE LOW.
- Ann. 215, Drop 41, FEEDPUMP SUCTION HEADER PRESSURE LOW.

Which ONE of the following describes the required operator actions?

A. Manually start the standby Hotwell Pump and then the Middle CBP.

B. Verify that the standby Hotwell Pump and the Middle CBP have both automatically started.

C. Manually start the Middle CBP and fully open CRV-224, Low Pressure Heater Bypass Valve.

D. Verify that the Middle CBP has automatically started and dispatch an operator to locally check CRV-224, Low Pressure Heater Bypass Valve position.

ANSWER: D

A. INCORRECT. The standby hotwell pump is NOT started and the middle CBP should auto start.
B. INCORRECT. The standby hotwell pump should NOT start and the middle CBP should AUTO start.
C. INCORRECT. The middle CBP should AUTO start and CRV-224 does NOT need to be open if sufficient CBPs are running.
D. CORRECT. The middle CBP should auto start and CRV-224 should be checked locally to verify position.

OBJECTIVE: RO-C-05400-E8; RO-C-05400-E3
REFERENCE: SOD-05500-001; 2-OHP-4024-215, Drops 41; 2-OHP-4024-216,
Drops 72 & 73.

KA - 056000 A2.04
Condensate System
Ability to (a) predict the impacts of the following malfunctions or operations on the Condensate System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Loss of condensate pumps
RO - 2.6 SRO - 2.8
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Question requires candidate to determine significance of alarms (including expected auto actions) and perform annunciator response (procedural) driven actions based on indications.

Original Question # - Modified from NRC EXAM 2008-093, 2008NRC-0419, NRC Exam 2007-90, 2007-0503, NRCAUDIT07-0935, INPO # 22973 Prairie Island -8/16/2002
Original Question KA - 056000 2.4.45

Changed question to address normal expected plant response due to a CBP trip. Removed failure of Middle CBP to start from the stem and changed distractors to focus on Manual or auto starting of the CBP and Hotwell pump. Removed FW pump trip options.

25. 025 007/RO/OK/DIRECT/NRC EXAM 2006-053-8/059000 A2.07/RO-C-05500-E5/NRCAUDIT07-0740/H/3

The following plant conditions exist:

- Unit 2 is at 9% power with the Main Turbine rolling at 1800 rpm.
- East Main Feedwater Pump (MFP) is supplying main feedwater to the Steam Generators (SG).
- Auxiliary Feed Water (AFW) pumps are aligned in Auto.
- All operating condensate booster pumps trip.

Which ONE of the following describes the system response and required operator action?

A. MFPs--Immediate Trip
MDAFW Pumps Start from MFP trip
Reactor Trip--On low low SG level
Enter OHP-4023-E-0, Reactor Trip or Safety Injection

B. MFPs--Immediate Trip
MDAFW Pumps Start & Turbine Trips from AMSAC
Enter OHP-4022-001-002, Loss of Load (Load Rejection)

C. MFPs--Trip after 5 sec. Delay
MDAFW Pumps Start from MFP trip
Reactor Trip--On low low SG level
Enter OHP-4023-E-0, Reactor Trip or Safety Injection

D. MFPs--Trip after 5 sec. Delay
MDAFW Pumps Start & Turbine Trips from AMSAC
Enter OHP-4022-001-002, Loss of Load (Load Rejection)

ANSWER: C

A. INCORRECT. The FW pumps will not trip immediately.
B. INCORRECT. The FW pumps will not trip immediately. AMSAC will not actuate since power is not >40%.
C. CORRECT. The MFPs will trip after FW suction pressure has lowered to <180 psig for 5 seconds. The trip of the Main FW pumps will cause the MDAFW pumps to start. The delayed trip of the FW pumps and the current power level will cause SG levels to Lower to the low low SG level reactor trip setpoint. After the reactor is tripped E-0 is entered.
D. INCORRECT. AMSAC will not actuate since power is not >40%.

OBJECTIVE: RO-C-EOP01-E1, RO-C-05500-E5, RO-C-05500-E10
REFERENCE: RO-C-05500, pgs 22 & 31; RQ-C-KNOW, pg 6 & 8

KA - 059000 A2.07
Main Feedwater (MFW) System
Ability to (a) predict the impacts of the following malfunctions or operations on the MFW System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Tripping of MFW pump turbine
RO - 3.0 SRO - 3.3
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Question tests ability to determine automatic actions that occur on a loss of MFW Turbine and the required operator actions/procedures.

Original Question # - NRC EXAM 2006-053-8, CM-7849, Master Bank 01055C0007-2
Original Question KA - 056000 A204, 035 A4.01

26. 026 006/RO/OK/DIRECT/RO25 AUDIT 055/061000 A1.04/RO-C-05600-E6/NRCAUDIT07-1025/H/2

Given the following:

- Unit 1 experienced a reactor trip from 100% power.
- The crew is currently taking actions in 1-OHP-4023-ES-0.1, Reactor Trip Response.
- Steam generator levels are being maintained at 40% with both Motor Driven AFW pumps.
- Unit 1 CST level is at 25% and rapidly lowering.
- Unit 2 CST is at 89% and stable.
- ESW headers are available to both units.
- The Makeup Water plant is available.

You have just received a call in the control room that a forklift has punctured the side of Unit 1 CST.

Based on the given conditions, if CST level lowers to ____ then the operator should align the AFW pumps suction to the alternate suction source starting with __________

A. 10%, Unit 1 Hotwell supply.

B. 10%, Makeup Water Plant.

C. 15%, the Emergency ESW supply.

D. 15%, the opposite Unit CST.

ANSWER: D

A. INCORRECT Switchover criteria is at 15%. Unit 2 CST is the preferred first source for these criteria.

B. INCORRECT Switchover criteria is at 15%. Unit 2 CST is the preferred first source for these criteria.

C. INCORRECT Unit 2 CST is the preferred first source for these criteria.

D. CORRECT Per the foldout page for ES-0.1, Switchover to alternate suction source for AFW should begin at 15% in the CST. If the CST is determined to NOT be intact then the operator is directed to go straight to CST crosstie with the opposite unit.

OBJECTIVE: RO-C-EOP03-E19, RO-C-05600-E6, RO-C-ES0-1-E4
REFERENCE: 1-OHP-4021-ES-0.1 (Foldout Page), 1-OHP-4022-055-003

KA - 061000 A1.04
Auxiliary / Emergency Feedwater (AFW) System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW System controls including:
AFW source tank level
RO - 3.9 SRO - 3.9
CFR - 41.5 / 45.5

KA Justification - Question requires operator to display knowledge of the CST/AFW suction limits and actions to prevent exceeding the CST/AFW suction limits.

Original Question # - RO25 AUDIT 055, NRCAUDIT07-1025, CM-8533
Original Question KA - 061000 A1.04

27. 027 009/RO/OK/DIRECT/RO24 AUDIT-049/062000 K1.03/RO-C-08200-E3/NRCAUDIT07-0379/F/2

The following plant conditions exist:

- 100% power.
- No equipment out of service.
- The Unit Auxiliary Transformers are supplying all plant equipment.
- An operator noted that the closed light for 1A7, Normal Feed Breaker to Bus 1A, was NOT lit.
- The light bulb was verified as good.

Which ONE of the following statements describes the condition for this breaker?

A. Breaker 1A7 cannot be remotely opened with the control switch.

B. An overload condition will cause breaker 1A7 to trip open.

C. A generator trip will cause breaker 1A7 to trip open.

D. Breaker 1A7 cannot be locally tripped.

ANSWER: A

A. CORRECT With the Close Light extinguished a loss of DC control power is indicated. This will prevent breaker operations with the control switch.

B. INCORRECT Plausible since some of the lower voltage breakers will trip on an overload even without DC control power. This breaker requires DC control to operate the trip solenoid and will not trip from overload.

C. INCORRECT Plausible is the operator has a misconception with breaker operations. This breaker requires DC control to operate the trip solenoid and will not open even though it receives a trip signal.

D. INCORRECT Plausible is the operator has a misconception with breaker operations. The breaker can be locally opened by operating the breaker trip pushbutton.

OBJECTIVE: RO-C-08200-E3
REFERENCE: RO-C-08200, Balance Of Plant Electrical System pg. 54

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 knowledge of the impact to the AC breaker due to a loss of DC control Power.

Original Question # - RO22-BOTH-37 (#34), RO24 AUDIT-049, NRCAUDIT07-0379, CM-7741
Original Question KA - 063.K3.02, 063000 - K1.02

28. 028 002/RO/OK/DIRECT/NRC EXAM 2007-57/SF6.063.A2.01/RO-C-AOP0550412-E1/2007-0522/H/4

The following plant conditions exist:

- Unit 2 is at 100% power, steady state conditions.
- A POSITIVE 250V ground exists on DC Bus 2CD.

If a NEGATIVE 250V ground also occurs on Bus 2CD, which ONE of the following describes the Plant response and the required operator actions? (Assume ground is on the bus bar)

A. The DC bus fuses will blow causing a complete loss of DC 2CD busses resulting in a Reactor Trip.
Perform actions of OHP-4023-E-0, OHP-4023-ES-0.1 and OHP-4022-082-002CD to stabilize the plant.

B. The Positive and Negative ground will balance out the circuit, however many relays will actuate causing a Reactor Trip.
Perform actions of OHP-4023-E-0, OHP-4023-ES-0.1 and OHP-4022-082-002CD to stabilize the plant.

C. The DC bus fuses will blow causing a complete loss of DC 2CD busses.
The Reactor will NOT Trip.
Perform actions of OHP-4022-082-002CD to stabilize the plant.

D. The Positive and Negative ground will balance out the circuit, however many relays will fail to actuate if required.
The Reactor will NOT Trip.
Perform actions of OHP-4022-082-002CD and begin a Unit shutdown.

ANSWER: A

A. CORRECT	The complete round on opposite busses will cause the fuse to blow and the loss of DC Bus 2CD. The Loss of either DC bus causes a reactor trip due to the RCP Under Frequency. [Note grounds of the same magnitude will typically cause the ground detection circuit to null out. This is true for all size grounds, but full grounds will also cause excessive current flow (similar to a direct short) which will result in a blown fuse.]
B. INCORRECT	The Grounds will cause a loss of DC Busses.
C. INCORRECT	The Reactor will trip.
D. INCORRECT	The Reactor will trip & the Grounds will cause a loss of DC Busses.

OBJECTIVE: RO-C-AOP0550412-E3, RO-C-AOP0550412-E1
REFERENCE: RO-C-08204 pgs. 11, 21-22,SD-08204 pg. 37, & RO-C-AOP13 Slide 26
KA - 063000 A2.01
D.C. Electrical Distribution System
Ability to (a) predict the impacts of the following malfunctions or operations on the D.C. Electrical System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Grounds
RO - 2.5 SRO - 3.2
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Question requires operator to determine impact of large grounds and then determine correct procedures to stabilize the plant.

Original Question #- NRC Exam 2007-57, 2007-0522, CM-40250
Original Question KA - 063000 A2.01

29. 029 003/RO/OK/DIRECT/RO27 AUDIT -50/SF6.064.K6.08/RO-C-03201-E8/RO27AUDIT-50/F/3

If the Emergency Diesel Generator (EDG) Day Tank level is unexpectedly lowering, which ONE of the following is the LOWEST of the listed quantities of fuel oil remaining in the Day Tank that will maintain the EDG OPERABLE per Technical Specifications?

A. 430 gallons.

B. 210 gallons.

C. 110 gallons.

D. 60 gallons.

ANSWER: C

A. INCORRECT Plausible since this number is above high day tank level.
B. INCORRECT Below the low air pressure alarm setpoint.
C. CORRECT TS requires 101.4 gallons (Low alarm 103~).
D. INCORRECT Plausible, since this normal frequency of the EDG.

OBJECTIVE: RO-C-03201-E8
REFERENCE: Technical Specification SR 3.8.1.4

KA - 064000 K6.08
Emergency Diesel Generator (ED/G) System
Knowledge of the effect of a loss or malfunction of the following will have on the ED/G System:
Fuel oil storage tanks
RO - 3.2 SRO - 3.3

KA Justification - Requires knowledge of the minimum level in the EDG fuel oil day tank that would allow the EDG to remain OPERABLE. CFR - 41.7 / 45.7

Original Question # - RO27 Audit -50, WattsBarMay2009-50
Original Question KA - 064000 K6.08

30. 030 005/RO/OK/DIRECT/NRC EXAM 2002-076-1/071000 A4.10/RO-C-02300-E3/NRCAUDIT07-0491/H/3

The in-service gas decay tank is being switched to another tank.
At the Waste Disposal System (WDS) panel you receive the Annunciator Panel 128 Drop 28, AUTO GAS ANALYZER, alarm.

A few minutes later you receive the following Annunciator Panel 128 alarms:

- Drop 10, WASTE GAS ANALYZER OXYGEN HIGH
- Drop 15, WASTE GAS ANALYZER O2 EXT HIGH

Drop 28 occurred ______, and Drops 10 and 15 occurred _______.

A. during the GDT tank transfer;
because there is high O2 in the in-service tank.

B. because the analyzer is removed from service before the tanks are switched;
due to the analyzer being placed back in service.

C. after the GDT tank transfer;
due to the analyzer being placed back in service.

D. because the analyzer is removed from service before the tanks are switched;
because there is high O2 in the in-service tank.

ANSWER: A

A. CORRECT	Per Panel 128 Drop 28 alarm response this alarm is expected during component position changes such as switching in-service GDTs. Drop 10 and Drop 15 will alarm when high O2 is sensed in the in-service GDT which has been aligned per 12-OHP-4021-023-001, Operation of the Waste Gas System.
B. INCORRECT	The gas analyzer is not removed from service prior to switching tanks.
C . INCORRECT	The gas analyzer is not removed from service prior to switching tanks.
D . INCORRECT	The gas analyzer is not removed from service prior to switching tanks.

OBJECTIVE: RO-C-02300-E3
REFERENCE: 1-OHP-4024-128, Annunciator #128 Response: Boron Recycle and Gas Waste, Drops 10, 15, and 28

KA - 071000 A4.10
Waste Gas Disposal System (WGDS)
Ability to manually operate and/or monitor in the control room:
WGDS sampling
RO - 2.5 SRO - 2.4
CFR - 41.7 / 45.5 to 45.8

KA Justification - Questions tests knowledge of the Waste Gas sampling system alarms that may be generated during a tank swap over.

Original Question # - NRCAUDIT07-0491, COOK NRC Exam 2002-076-1, 19434, RO23 Audit-118-3
Original Question KA -071000 - 2.4.46

31. 031 002/RO/OK/MODIFIED/NRC EXAM 2006-049-31/072000 A1.01/RO-C-02800-E9/NRCAUDIT07-0738A/H/3

The following conditions exist:

- Unit 1 was in Mode 1.
- A Containment Pressure Relief was in progress with the following lineup:
  VCR-107, Cntmt Press Relief Valve IC - OPEN
  VCR-207, Cntmt Press Relief Valve OC - OPEN
  HV-CPR-1, CNTMT Press Relief Fan - RUNNING
- A HIGH alarm on VRS-1101, Upper Containment Normal Range Monitor, occurs.

Which ONE of the following describes the required operator response for the Containment Pressure Relief System due to the failure alarm?

A. Manually close VCR-107.
Verify VCR-207 has automatically closed.
Manually trip HV-CPR-1.

B. Verify VCR-107 has automatically closed.
Manually close VCR-207.
Manually trip HV-CPR-1.

C. Verify VCR-107 has automatically closed.
Verify VCR-207 has automatically closed.
Verify HV-CPR-1 has automatically tripped.

D. Verify VCR-107 has automatically closed.
Verify VCR-207 has automatically closed.
Manually trip HV-CPR-1.

ANSWER: B

A. INCORRECT	This is partially correct for VRS-1201 (except fan auto trips).
B. CORRECT	Correct Actions for failure of the VRS-1101. VCR-107 is closed by the VRS-1101.
C. INCORRECT	Only VCR-107 is closed by the VRS-1101 channel. This would be the response if both channels (1101/1201) sensed a high radiation condition.
C. INCORRECT	Only VCR-107 is closed by the VRS-1101 channel.

OBJECTIVE: RO-C-02800-E9
REFERENCE: SOD-1350-001; 12-OHP-4024-139, Drop 1, 1-OHP-4021-028-004.

KA - 072000 A1.01
Area Radiation Monitoring (ARM) System
Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including:
Radiation levels
RO - 3.4 SRO - 3.6
CFR - 41.5 / 45.5

KA Justification - Question tests ability to predict radiation monitor actions associated with radiation levels.

Original Question # - Modified from 2006 NRC EXAM-049-31, NRCAUDIT07-0738
Original Question KA - 029000 2.4.31

Modified by changing stem from VRS1201 to VRS 1101, which changes correct answer to B vs. A. Also Modified Distractor D for 1101/1201.

32. 032 005/RO/OK/DIRECT - REPEAT/NRC EXAM 2010-73/073000 2.4.49/RO-C-02800-E9/NRCAUDIT07-0583/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 High 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 VCR-101 through VCR-106 close;
- HV-CPS-1/2, Containment Purge Supply Fans 1 and 2, trip;
- HV-CPX-1/2, Containment Purge Exhaust Fans 1 and 2, trip;
- HV-CPR-1, Containment Pressure Relief Fan, trips;
- HV-CIPS-1, Containment Instrument Room Purge Supply Fan, trips.

D. Verify the following:
- Containment ventilation isolation valves VCR-201 through VCR-206 close;
- HV-CPS-1/2, Containment Purge Supply Fans 1 and 2, trip;
- HV-CPX-1/2, Containment Purge Exhaust Fans 1 and 2, trip;
- HV-CPR-1, Containment Pressure Relief Fan, trips;
- 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.

OBJECTIVE: RO-C-02800-E9
REFERENCE: 1-OHP-4021-028-005, Att. 2
KA - 073000 2.4.49
Process Radiation Monitoring (PRM) System
Emergency Procedures/Plan
Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
RO - 4.6 SRO - 4.4
CFR - 41.10 / 43.2 / 45.6

KA Justification - Question tests ability to identify conditions requiring manual termination of Containment Purge.

Original Question # - NRC 2010 Exam-73, NRC Exam 2004-049-5, AUDIT02-BOTH31, NRCAUDIT07-0583
Original Question KA - 194001 2.3.11, 2.3.9

33. 033 003/RO/OK/DIRECT/RO25 AUDIT-72/073000 K5.03/RO-C-RP02-E5/NRCAUDIT07-1007/H/3

Units 1 and 2 are at 100% power.

The following conditions exist:

- Unit 1 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 300 mrem/hr.
- In order to reach the location of the repair the worker must transit through a 8 Rem/hr high radiation area for 3 minutes and return via the same path.
- The worker currently has a Cook Plant accumulated annual dose of 200 mrem.

Which ONE of the following is the maximum allowable time that the worker can participate in the repairs and NOT exceed the Site TEDE Administrative Dose Limit?

A. 200 minutes

B. 280 minutes

C. 360 minutes

D. 400 minutes

ANSWER: A

A. CORRECT	The candidate should determine that the ADL is 2000 mrem. Transient exposure is 800 mrem (8000mrem/hr x 6/60hr). (transit to and from the job). (Current) 200 mrem + (transit) 800 mrem = 1000 mrem ADL of 2000 mrem - 1000 mrem = 1000 mrem allowable before reaching ADL. 1000 mrem /300 mrem/hr = 3.33 hours
B. INCORRECT	Based on calculating using a one-way transit dose.
C. INCORRECT	Based on using ADL (2000) and NO transit dose.
D. INCORRECT	Based on using a limit of 3000 versus correct ADL (2000).

OBJECTIVE: RO-C-RP02-E4, RO-C-RP02-E5
REFERENCE: RO-C-RP02

KA - 073000 K5.03
Process Radiation Monitoring (PRM) System
Knowledge of the operational implications of the following concepts as they apply to the PRM System:
Relationship between radiation intensity and exposure limits
RO - 2.9 SRO - 3.4
CFR - 41.5 / 45.7

KA Justification - Question requires candidate to determine exposure time based on limits and radiation intensity measured in the plant.

Original Question # - RO25 AUDIT-72, NRCAUDIT07-1007, Modified From NRC EXAM 2004-099-3
Original Question KA - Generic 2.3.1

34. 034 003/RO/OK/DIRECT/NRC EXAM 2007-63/076000 K3.01/RO-C-01900-E5/CM-7744A/H/3

Given the following:

- U1 'W' ESW Pump is Running.
- U2 'W' ESW Pump is Running.
- U1 'E' ESW Pump is in Standby.
- U2 'E' ESW Pump is in Standby.

If the U2 'W' ESW Pump motor fails, the _______ will be supplied with cooling water from the _______.

A. 2E CCW Hx, 2E ESW pump

B. 2E CCW Hx, 1E ESW pump

C. 2W CCW Hx, 2E ESW pump

D. 2W CCW Hx, 1E ESW pump

ANSWER: D

A. INCORRECT Plausible if the candidate thinks that the 2E pump will auto start on loss of 2W ESW pump, but the ESW system is cross connected between units.

B. INCORRECT Plausible if the candidate thinks that the 2E pump supplies the unit 2 east train, but the ESW system is cross connected between units.

C. INCORRECT Plausible if the candidate thinks that the 2E pump will auto start on loss of 2W ESW pump, but the ESW system is cross connected between units.

D. CORRECT The ESW System normally operates with two of four ESW pumps running, one pump supplying a header on each unit. Either the Unit 1 East or Unit 2 West Pumps supply the Unit 1 East Header and the Unit 2 West Header. Either the Unit 1 West or Unit 2 East Pump supplies the Unit 1 West Header and the Unit 2 East Header. A low ESW header pressure at 40 psig will start the standby pump (if in auto).

LESSON PLAN/OBJ: RO-C-01900-E2, RO-C-01900-E5
REFERENCE: SOD-01900-001

KA - 076000 K3.01
Service Water System (SWS)
Knowledge of the effect that a loss or malfunction of the SWS will have on the following:
Closed cooling water
RO - 3.4 SRO - 3.6
CFR - 41.7 / 45.6

KA Justification - Question tests knowledge of the effect that a Loss of ESW (SWS) has on the CCW system.

Original Question # - CM-7744, CM-0247, 2007-0548, NRCAUDIT07-0971, NRC EXAM 2007-63, 01019C0005-5, RO23-073-3 (#067), RO24-053-9
Original Question KA - 076000 K4.06

35. 035 012/RO/OK/MODIFIED/NRC EXAM 2007-64/078000 A3.01/RO-C-06401-E4/RO26-0116/F/2

The Plant and Control Air Systems are aligned as follows:

- Unit 1 Plant Air Compressor (PAC) is running loaded.
- Unit 2 PAC is in standby (Auto) alignment.
- Both Control Air Compressors (CACs) are in standby (Auto) alignment.

The following events occur:

- Unit 1 PAC trips.
- Unit 1 "PAC OVERLOAD TRIP" alarm annunciates.

If air header pressure drops continuously, in what order will the following automatic actions occur?

1) Plant Air Header Crosstie Valves CLOSE
2) Control Air Compressors (CACs) START
3) U-2 Plant Air Compressor (PAC) STARTS

A. 1, 2, 3

B. 1, 3, 2

C. 3, 2, 1

D. 3, 1, 2

ANSWER: C

A. INCORRECT. It would be plausible for the air header to isolate any potential leakage paths prior to starting the compressors.
B. INCORRECT. It would be plausible for the air header to isolate any potential leakage paths prior to starting the compressors.
C. CORRECT. The PAC will start first and attempt to restore pressure, if it fails the CAC will start, if that also fails the air header will isolate to section off the leak.
D. INCORRECT. It would be plausible for the PAC to start and then attempt to isolate leakage paths prior to starting the CAC.

Note: Per SOD-06401-002, Plant Air System Drawing, the following are the Air Pressure Setpoints:

Air Header Pressure (psig):

125	Safeties open
104	PAC surge unloader opens
100	CAC unloads
98	PA header unisolates
97	PA alarm ‘PAC fail/low press’
95	STANDBY PAC starts(3)
95	CA low press alarm
90	CAC auto start(2)
85	Plant air header isolates(1)
80	Manual reactor trip

OBJECTIVE: RO-C-06401-E4
REFERENCE: SOD-06401-002

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 addresses automatic actions that are expected with the Plant & Control (Instrument) Air systems.

Original Question # - RO26-0116, RO 26 AUDIT-56, modified from NRC EXAM 2007-64, INPO # 30321 Point Beach -2006
Original Question KA - 078000 A3.01

Modified question for RO26 Audit by moving Alarm to the stem from the choices and re-arranging distractor order making C the correct answer.

36. 036 003/RO/OK/DIRECT/NRC EXAM 2006-21/086000 K3.01/RO-C-EC02-E4/NRCAUDIT07-1018/H/3

The control room has been evacuated per 2-OHP-4025-001-001, Emergency Remote Shutdown.

The crew has performed 2-OHP-4025-LS-6-1, Seal Injection from CVCS Cross-Tie.

Which ONE of the following describes the source and method of monitoring borated water addition?

Prior to aligning RHR for shutdown cooling, RCS boration is accomplished by ...

A. manually aligning Unit 1 Boric Acid makeup to feed Unit 2 through the crosstie. Flow is tracked by Unit 1 Boric Acid flow recorder.

B. locally aligning Unit 2 Boric Acid makeup to feed Unit 2 through the crosstie. Flow is tracked by Boric Acid Flow indicator on the Local Instrument Panel.

C. manually aligning Unit 1 CCP suction to the RWST to feed Unit 2 through the crosstie. Flow is tracked by the sum of local seal injection flows added to the Unit 1/2 CCP Crosstie Flow indicator.

D. manually aligning Unit 1 CCP suction to the RWST to feed Unit 2 through the crosstie. Flow is tracked by the Unit 1/2 CCP Crosstie Flow indicator.

ANSWER: D

A. INCORRECT	Unit 1 is aligned to the RWST. If Unit 1 was aligned to the VCT, this would be a valid flowpath.
B. INCORRECT	Unit 2 CCPs are placed in Lock-out prior to evacuating the control room. Unit 2 Boric Acid flow would be at too low of pressure to inject into the discharge pressure of the Unit 1 CCP through the crosstie. The Pressurizer level is monitored at the local panels.
C. INCORRECT	All Unit Crosstie flow travels through the Unit 1/2 CCP Crosstie Flow indicator. Adding the seal injection flow would count that flow twice.
D. CORRECT	Prior to initiating the Unit Crosstie, Unit 1 is directed to trip the unit and align CCP suction to the RWST. The Total flow from Unit 1 to Unit 2 is indicated on the Unit 1/2 CCP Crosstie Flow indicator.

OBJECTIVE: RO-C-EC02-E4
REFERENCE: 2-OHP-4025-001-001, Emergency Remote Shutdown, pg 10;
2-OHP-4025-LS-6-1, pgs 3-8

KA - 086000 K3.01
Fire Protection System (FPS)
Knowledge of the effect that a loss or malfunction of the Fire Protection System will have on the following:
Shutdown capability with redundant equipment
RO - 2.7 SRO - 3.2
CFR - 41.7 / 45.6

KA Justification - Question tests knowledge of actions required outside of the control in the event the Fire Protection System did not function to protect controls.

Original Question #- NRCAUDIT07-1018, NRC Exam 2006-21
Original Question KA - 086000 K3.01

37. 037 005/RO/OK - ATTACHMENT/DIRECT/RO24 AUDIT-001-12/103000 2.4.1/RO-C-EOP13-E4/NRCAUDIT07-0774/H/3

The control room crew has just completed immediate actions of 2-OHP-4023-E-0, Reactor Trip or Safety Injection and completed transition to 2-OHP-4023-E-1, Loss Of Reactor Or Secondary Coolant.

The Balance of Plant Operator notes Critical Safety Function Status as follows:
Subcriticality - GREEN
Core Cooling - YELLOW
Heat Sink - GREEN
Integrity - YELLOW

The Shift Manager checks containment conditions to determine if the containment barrier is intact. The following conditions exist:

- Containment pressure: 8 psig and slowly rising
- Containment water level "Flood Level" lights: LIT
- Containment Radiation: 8.0 R/hr and stable
- All Containment Phase A & B penetrations are isolated

Given these conditions, what is the correct crew response?

Attachment Provided

A. Transition to 2-OHP-4023-FR-Z.2, Response to Containment Flooding

B. Transition to 2-OHP-4023-FR-Z.1, Response to High Containment Pressure

C. Remain in 2-OHP-4023-E-1, Yellow Path Functional Restoration Procedures are entered on discretion only.

D. Remain in 2-OHP-4023-E-1, until first transition at which time Functional Restoration Procedures become applicable.

ANSWER: B

A. INCORRECT	Containment Flooding Orange Path is a lower priority than Containment Pressure Orange Path. Candidate may not know that Containment Pressure Orange path also uses FR-Z.1.
B. CORRECT	Containment conditions generate an ORANGE path containment CSF status tree. Per rules of usage an orange path requires transition to appropriate FRP. In this case FR-Z.1 is the correct procedure.
C. INCORRECT	An orange path exists and rules of usage require transition, the second part of distractor is plausible because it is a true statement.
D. INCORRECT	An orange path exists and rules of usage require transition. Plausible if candidate thinks transition is delayed until after OHP-4023-E-1 versus when the first transition from OHP-4023-E-0 occurred.

OBJECTIVE: RO-C-EOP01-E22, RO-C-EOP13-E4
REFERENCE: OHP-4023-F-0.5 Status Tree; OHI-4023 pgs 10 & 51

Attachment Provided - OHP-4023-F-0.5 Containment Status Tree

KA - 103000 2.4.1
Containment System
Emergency Procedures/Plan
Knowledge of EOP entry conditions and immediate action steps.
RO - 4.6 SRO - 4.8
CFR - 41.10 / 43.5 / 45.13

KA Justification - Question requires operator to determine the correct EOP entry required based on Containment conditions.

Original Question # - RO24 AUDIT-001-12
Original Question KA - 1030002401

38. 038 003/RO/OK/DIRECT/RO25 AUDIT-37/103000 A1.01/RO-C-03400-E12/NRCAUDIT07-1073/F/3

A malfunction of the Containment Chilled Water (CHW) system has resulted in a partial loss of CHW flow to the Containment Ventilation Units.

Under this condition, which ONE of the following sets of containment readings would result in **Unit 2** containment still being within **Technical Specification** limits?

A. Pressure is 0.18 PSIG,
Upper containment temperature 97°F,
Lower containment temperature 110°F.

B. Pressure is 0.35 PSIG,
Upper containment temperature 105°F,
Lower containment temperature 97°F.

C. Pressure is 0.15 PSIG,
Upper containment temperature 105°F,
Lower containment temperature 105°F.

D. Pressure is -1.55 PSIG,
Upper containment temperature 97°F,
Lower containment temperature 115°F.

ANSWER: A

A - CORRECT. LCO 3.6.4 Containment pressure shall be $\geq$ -1.5 psig and $\leq$ +0.3 psig. LCO 3.6.5 Containment average air temperature shall be:
a. $\geq$60°F and $\leq$ 100°F for the containment upper compartment and
b. $\geq$60°F and $\leq$ 120°F for the containment lower compartment.

B - INCORRECT. Pressure too high & Upper temp too high, plausible since upper temp is below lower temp limit & Pressure is still reasonable.

C - INCORRECT. Upper Temp too high. Plausible if candidate doesn't know that Upper limit is lower.

D - INCORRECT. Pressure too low. Plausible since temps are in range and pressure is usually a concern only for higher temps.

OBJECTIVE: RO-C-03400-E12
REFERENCE: Unit 2 T.S. 3.6.4 and 3.6.5

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

KA Justification - This meets the K/A because the candidate must know the Containment limits that are being monitored.

Original Question # - NRCAUDIT07-1073, RO23-090-4-Q72, RO25 Audit-37
Original Question KA - SYS022 A1.04

39. 039 005/RO/OK/NEW/NEW/APE.008.AK1.02/RO-C-EOP02-E3/RO-C-EOP02-E3-1/H/3

Given the following conditions:

- Unit 1 tripped from 100% power.
- Pressurizer (PRZ) PORV NRV-151 opens and sticks partially open.
- The associated PORV Block Valve could NOT be closed.
- Pressurizer Relief Tank (PRT) pressure rose to the point that the Rupture Disc ruptured.
- Safety Injection actuated and filled the pressurizer (water solid).
- The crew is currently performing OHP-4023-ES-1.2, Post LOCA Cooldown and Depressurization.

The PRZ PORV leakage was estimated at 100 gpm when the Reactor Coolant System (RCS) pressure was at 2000 psig with PRT pressure reading 3 psig. (assume water solid conditions)

Which ONE of the following is the approximate current leak rate if RCS pressure is 1000 psig with PRT pressure reading 3 psig?

Assume: NRV-151 position has not changed.

A. Approximately 25 gpm

B. Approximately 50 gpm

C. Approximately 70 gpm

D. Approximately 100 gpm

ANSWER: C

A. INCORRECT	Differential pressure is 1/2 of original and this flow is the DP squared.
B. INCORRECT	This is the value for 50% of the flow since DP is 1/2.
C. CORRECT	Break flow is Proportional to the Square Root of the Pressure Differential. Flow is ~70% of initial or 70 gpm Flow Int. = 100GPM ~ Square Root (2000-3) = 44.67 Flow FINAL = X GPM ~ Square Root (1000-3) = 31.57 Flow FINAL = X GPM = 31.57/44.67 times 100 GPM = 70.7 gpm
D. INCORRECT	This is original value. The DP changed and so does break flow.

OBJECTIVE: RO-C-GF27-E10; RO-C-EOP02-E3
REFERENCE: RO-C-GF27 (PC07I, pg 90)

KA - 000008 AK1.02
Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)
Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident:
Change in leak rate with change in pressure
RO - 3.1 SRO - 3.7
CFR - 41.8 / 41.10 / 45.3

KA Justification - Question requires operator to determine how leak flow changes as the pressure changes.

Original Question # - NEW NRC EXAM 2012-039
Original Question KA - N/A

40. 040 001/RO/OK/NEW/NEW/EPE.009.EA2.24/RO-C-EOP03-E2/RO-C-EOP03-E2-1/H/3

Unit 2 is responding to a Small Break LOCA in accordance with OHP-4023-ES-1-2, Post LOCA Cooldown and Depressurization. Step 14 directs the operators to check if an Reactor Coolant Pump (RCP) should be started.

Given the following indications:

- Panel 207, Drops 14, 34, 74, & 94, RCP SEAL OUTLET TEMP HIGH alarms - LIT.
- No. 1 seal leakoff temperatures are 208°F.

Which ONE of the following best describes the restoration or non-restoration of RCP seal cooling and the requirements for RCP start?

A. Slowly restore seal cooling, limiting the cooldown rate to 1°F/minute. Do NOT start the RCP until seal leakoff temperatures are less than 200°F.

B. Do NOT restore seal cooling. If seal leakoff flow is acceptable, the RCP may be used for a forced circulation cooldown and depressurization of the plant.

C. Restore seal cooling as soon as possible to minimize the potential for seal degradation. The RCP should be started to minimize the potential for shaft warpage.

D. Do NOT restore seal cooling due to potential damage from thermal shock to the RCP seals. Do NOT start the RCP until the Reactor Coolant System (RCS) has cooled the seals and an evaluation is completed.

ANSWER: D

A. INCORRECT Plausible since old procedure guidance restored seal cooling in this manner. Also seals must be cooled slowly but this is accomplished by cooling the RCS. No. 1 seal leakoff temperature > 200°F is RCP trip criteria.

B. INCORRECT Plausible since the pumps may be run in other EOPs without normal conditions and the acceptable seal leakoff is part of the status evaluation.

C. INCORRECT Plausible since it is normally desired to restore seal injection as soon as possible and start the pump to prevent warpage from uneven cooling. Rapidly restoring seal injection could cause seal failure.

D. CORRECT If all seal cooling has been lost long enough that the maximum RCP seal parameters identified in the RCP Vendor Manual have been exceeded, seal injection and CCW thermal barrier cooling should not be re-established to the affected RCP(s). Seal cooling should instead be restored by cooling the RCS, which will reduce the temperature of the water flowing through the pump seals.

OBJECTIVE: RO-C-EOP03-E2
REFERENCE: 2-OHP 4023.SUP.007, pg 2; 12-OHP 4023.SUP.007 PSBD, pg 4

KA - 000009 EA2.24
Small Break LOCA
Ability to determine and interpret the following as they apply to a small break LOCA:
RCP temperature setpoints
RO - 2.6 SRO - 2.9
CFR - 43.5 / 45.13

KA Justification - Question requires operator to determine correct actions concerning RCP start during a small break LOCA based on the seal temperature.

Original Question #- NEW NRC Exam 2012-040
Original Question KA - N/A

41. 041 003/RO/OK/NEW/NEW/000011 EK3.13/RO-C-EOP09-E4/RO-C-EOP09-E4-1/F/2

After a large break LOCA, recirculation is transferred from cold leg recirculation to hot leg recirculation.

Which ONE of the following is the **primary reason** hot leg recirculation is used at this point rather than continuing to use cold leg recirculation?

A. To prevent boron precipitation in the core.

B. To cool the upper head and reduce thermal stresses.

C. To flush any "hard bubble" of noncondensible gasses out of the head.

D. To collapse any voids that have formed on the coolant side of the steam generator tubes.

ANSWER: A

A. CORRECT The transfer to hot leg recirculation is made to prevent the plate out of boron on heat transfer surfaces.
B. INCORRECT RCS & vessel head temperatures should be stabilized at this time.
C. INCORRECT Hard bubbles will need to be vented, not flushed out.
D. INCORRECT Voids are collapsed but in the total RCS not just in the SG tubes.
OBJECTIVE: RO-C-EOP09-E4; RO-C-EOP09-E36
REFERENCE: 12-OHP-4023-ES-1.4 PSBD, pg 4

KA - 000011 EK3.13
Large Break LOCA
Knowledge of the reasons for the following responses as they apply to the Large Break LOCA:
Hot-leg injection/recirculation
RO - 3.8 SRO - 4.2
CFR - 41.5 / 41.10 / 45.6 / 45.13

KA Justification - Question requires operator knowledge of the reason for hot leg recirculation.

Original Question #- NEW NRC EXAM 2012-041
Original Question KA - N/A

42. 042 006/RO/OK/DIRECT/NRC EXAM 2002-052/000015 AK2.08/RO-C-AOP0420412-E3/NRCAUDIT07-0820/H/3

Simultaneous faults on BOTH T11A & T11D Buses at 100% Power requires an immediate...

A. reactor trip because the RCP bearings will overheat without Component Cooling flow.

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

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

D. 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.

OBJECTIVE: RO-C-AOP0420412-E3, RO-C-01600-E5
REFERENCE: RO-C-01600, PG 14; 1-OHP-4022-016-004, Loss of Component Cooling Water, pg 2

KA - 000015 AK2.08
017 Reactor Coolant Pump (RCP) Malfunctions
Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions and the following:
CCWS
RO - 2.6 SRO - 2.6
CFR - 41.7 / 45.7

KA Justification - Questions requires the operator to determine that a loss of power will cause a loss of CCW and the impact on the RCPS and plant.

Original Question # - NRCAUDIT07-0820, NRCAUDIT07-0189, INPO - DIRECT 20154, COOK02-052-1, RO24 Audit-47-12
Original Question KA - 003000-K2.02, 062.A2.01

43. 043 006/RO/OK/NEW/NEW/000025 AA1.08/RO-C-01700-E4/RO-C-01700-E6-1/H/3

Given the following conditions:

- Unit 1 is in Mode 4 during cooldown per OHP-4021-001-004, Plant Cooldown from Hot Standby to Cold Shutdown.
- West Residual Heat Removal (RHR) Pump and Heat Exchanger are operating, aligned to the cooldown path through injection lines to Cold Legs Loops 2 & 3.
- Reactor Coolant System (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 RCS and Component Cooling Water (CCW) system?

A. The RCS Cooldown rate and CCW temperatures will both LOWER.

B. The RCS Cooldown rate will RISE and CCW temperatures will LOWER.

C. The RCS Cooldown rate and CCW temperatures will both RISE.

D. The RCS Cooldown rate and CCW temperatures will both REMAIN THE SAME.

ANSWER: C

A. INCORRECT	IRV-320 fails open so flow will rise. Plausible if operator assumes that IRV-320 fails closed.
B. INCORRECT	IRV-320 fails open so flow will rise. This will raise the RCS Cooldown rate. Plausible if confusion over impact of Heat exchanger flow rates on CCW.
C. CORRECT	IRV-320 fails open on loss of air. This will raise RHR flow through the HX causing the RCS return temperature to lower and more heat transfer to CCW.
D. INCORRECT	IRV-320 fails open on loss of air. Plausible if IRV-320 position didn't change, or if IRV-311 controlled in auto like mini flows.

OBJECTIVE: RO-C-01700-E4
REFERENCE: 1-OHP-4021-017-002, Placing in Service the RHR System;
1-OHP-4022-064-002, Loss of Control Air Recovery (Step 43 & Att. B-9),
SOD-01700-001

KA - 000025 AA1.08
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:
RHR cooler inlet and outlet temperature indicators
RO - 2.9 SRO - 2.9
CFR - 41.7 / 45.5 / 45.6

KA Justification - Question requires operator to determine the failure direction of HX flow control and the impact on RHR cooler inlet and outlet temperatures (HX DT & RHR & CCW outlet temps).

Original Question # - New 2012-043
Original Question KA - New

44. 044 002/RO/OK/DIRECT/RO25 AUDIT-007/000026 AK3.04/RO-C-AOP0420412-E3/CM-8485/H/3

After indications of a lowering Component Cooling Water (CCW) Surge tank level the crew entered 2-OHP-4022-016-001, Malfunction of the CCW System.

The Crew has started the West CCW pump, split the East and West Headers, and identified that the leak is downstream of 2-CMO-410, East CCW HX Outlet Valve.

The East CCW pump is shutdown and Attachment A is performed to isolate the CCW leak.

In this alignment, the West CCW pump will supply cooling to the West Safeguards Header and the ...

A. East Safeguards Header. The Miscellaneous Services Header will also be supplied through the crosstie.

B. Miscellaneous Services Header. The East Safeguards Header will be WITHOUT Cooling.

C. East Safeguards Header. The Miscellaneous Services Header will be WITHOUT Cooling.

D. Miscellaneous Services Header. The East Safeguards Header will be crosstied to Unit 1 CCW Cooling.

ANSWER: B

A. INCORRECT	The West CCW can NOT support the East CCW Header with the leak isolated.
B. CORRECT	The Miscellaneous Services Header is aligned to only the West CCW header when the trains are split. The East CCW header must be isolated to isolate the leak.
C. INCORRECT	The Misc Header was aligned to the West Header. The East header can NOT be aligned to the West CCW system.
D. INCORRECT	The Unit Crossties are Upstream of the HX.

OBJECTIVE: RO-C-AOP0420412-E3
REFERENCE: 2-OHP-4022-016-001, Malfunction of the CCW System Steps 5-11; RO-C-01600

KA - 000026 AK3.04
Loss of Component Cooling Water (CCW)
Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water:
Effect on the CCW flow header of a loss of CCW
RO - 3.5 SRO - 3.7
CFR - 41.5 / 41.10 / 45.6 / 45.13

KA Justification - Question requires the operator to determine which CCW headers will still have flow following isolation of a CCW leak.

Original Question # - CM-8485, RO25 AUDIT-007
Original Question KA - 000026 K3.04

45. 045 005/RO/OK/DIRECT/RO25 AUDIT-8/000027 AA2.03/RO-C-00202-E13/NRCAUDIT07-0978/H/4

The Unit is operating at 100% power with the following conditions:

- Pressurizer PRESS CTRL SELECTOR switch is in the Channel 1-2 position.
- NPP-153, Channel 3 Pressurizer Pressure, failed HIGH four minutes ago.
- RU-27, PRZ Pressure Controller has just failed to 100%.

Assuming NO operator actions, how will the failure of the Pressurizer Pressure Master Control to 100% demand affect pressurizer pressure?

A. No PORVs will open. Pressurizer heaters will energize. Pressure will rise to the reactor trip setpoint.

B. PORV NRV-152 remains CLOSED and NRV-153 and 151 will OPEN.
Pressure will lower to the backup heater setpoint. Pressurizer heaters will energize.

C. Pressurizer sprays will open. PORV NRV-152 will OPEN and NRV-153 and 151 will remain CLOSED. Pressure will lower to the reactor trip setpoint.

D. Pressurizer sprays will open. No PORVs will open. Pressure will lower to the reactor trip setpoint.

ANSWER: D

A. INCORRECT	This is the response for the Controlling Channel failing High or the controller failing Low.
B. INCORRECT	NRV-153 & 151 have arming signals from Channel 3 but not the open signal from Channel 2. Heaters will not energize since RU-27 is failed high.
C. INCORRECT	NRV-152 is Not armed (Channel 4). This would be correct if Channel 3 armed NRV-152.
D. CORRECT	The indications given represent a failure high of 2-RU-27, PRZ Press Control. This will cause the spray valves to open, the high pressure alarm and actual pressure to lower. The pressure master controls all heaters, pressurizer spray and PORV NRV-152 & but it is not armed (from Channel 4). With the RU 27 Demand at 100% the signal is sent to Open the Sprays fully (and PORV NRV-152 except it is not armed). This will result in Lowering PRZ Pressure. The heaters will not energize since the Signal from RU-27 remains high.

OBJECTIVE: RO-C-00202-E13
REFERENCE: SOD-00202-001 & SOD-202-002

KA - 000027 AA2.03
Pressurizer Pressure Control (PZR PCS) Malfunction
Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions:
Effects of RCS pressure changes on key components in plant
RO - 3.3 SRO - 3.4
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Question requires operator to determine the results of multiple failures within the Pressurizer Pressure control system and the resulting impact of the lowering RCS pressure on the plant.

Original Question # - 045
Original Question KA - 000027 AK2.03

46. 046 003/RO/OK/DIRECT/RO27 AUDIT -8/000029 EK2.06/RO-C-01200-E11/CM-7599A/H/3

Given the following condition:

- The crew is implementing OHP-4023-FR-S.1, Response to Nuclear Power Generation/ATWS

Which ONE of the following combinations of breaker positions indicate that the reactor has been tripped?

Note: The Motor Generator INPUT Breakers are both closed.

- RTA = Reactor Trip Breaker A
- RTB = Reactor Trip Breaker B
- BYA = Reactor Trip Bypass Breaker A
- BYB = Reactor Trip Bypass Breaker B
- MGN Output = MG North output Breaker
- MGS Output = MG South output Breaker

LEGEND: X = CLOSED; O = OPEN

	RTA	RTB	BYA	BYB	MGN	MGS
A.	X	X	O	O	X	O
B.	X	O	O	X	O	X
C.	O	X	X	O	X	X
D.	X	O	X	O	X	X

ANSWER: D

A - INCORRECT. Only one MG set breaker is open. Both required to be open to trip reactor.
B - INCORRECT. RTA and BYB closed will provide flowpath to the rod coils. Only one MG set breaker is open. Both required to be open to trip reactor.
C - INCORRECT. RTB and BYA closed will provide flowpath to the rod coils.
D - CORRECT. With both RTB and BYB open, the MG set output supply to the rod coils is interrupted, which will result in the rods dropping into the core (a reactor trip).

OBJECTIVE: RO-C-01200-E11
REFERENCE: SOD-01200-002, RO-C-1200 Slide 22

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 ability to monitor the MG set and Reactor Trip breakers during an ATWS to determine when the reactor has been successfully tripped.

Original Question # - RO27 Audit -8, MASTER BANK 012-11, CM-8640, CM-7599
Original Question KA - 000029 EA1.12

47. 047 004/RO/OK/MODIFIED/NRC EXAM 2008-66/000036 AK1.01/RO-C-AOP0630412-E3/NRCAUDIT07-1024A/H/3

Given the following plant conditions on Unit 2:

- The unit is in Mode 6 with refueling activities in progress.
- Containment purge is in service.
- A fuel element is accidentally dropped into the cavity.
- All radiation monitor TRIP/BLOCK switches are in their NORMAL position.
- The Manipulator Crane area radiation monitor has a HIGH radiation alarm.
- Containment Radiation Monitors show a rise but no alarms have actuated.

(Assume: All equipment responds as designed.)

Which ONE of the following actions would occur, assuming that operators follow the instructions of 12-OHP-4022-018-004, Irradiated Fuel Handling Accident In Containment Building - Control Room?

A. Verify containment evacuation alarm sounds automatically.
Verify containment purge stops automatically.

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

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

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

ANSWER: D

A. INCORRECT. Containment evacuation alarm is manually actuated by the control room operator. Containment purge will NOT stop automatically without Containment Monitors being in High alarm.

B. INCORRECT. Containment purge will NOT stop automatically without Containment Monitors being in High alarm.

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

D. CORRECT. 12-OHP-4022-018-004, Step 1 directs the operator to actuate the Containment Evacuation alarm. For the protection of personnel, it is important to evacuate the affected area until radiation surveys can be completed. Step 2 follows up the alarm with a page announcement notifying all non-essential personnel to evacuate the containment. Steps 5 through 9 verify the containment purge and pressure relief systems are shutdown and isolated. This will limit the exposure of personnel outside containment.

OBJECTIVE: RO-C-AOP0630412-E3
REFERENCE: 12-OHP-4022-018-004, pg 3

KA - 000036 AK1.01
Fuel Handling Incidents
Knowledge of the operational implications of the following concepts as they apply to Fuel Handling Incidents:
Radiation exposure hazards
RO - 3.5 SRO - 4.1
CFR - 41.8 / 41.10 / 45.3

KA Justification - Question tests candidates knowledge of actions required (to limit radiation exposure) per the procedure for an accident with irradiated fuel.

Original Question # - NRC Exam 2008-66, RO25 AUDIT- 21, RO-22 AUDIT BOTH-76 (#68), CM-8480, 2008NRC-0559, NRCAUDIT07-0319, NRCAUDIT07-1024

Original Question KA - APE036 G2.4.31, SYS 103 A2.04, 194001 2.1.42

Modify by removing ERS 2305 & 2405 in High Alarm from Stem - Removes automatic actions associated with these monitors. This changes correct answer to D.

48. 048 009/RO/OK - ATTACHMENT/NEW/NEW/000038 EK1.01/RO-C-05200-E9/RO-C-05200-E9-1/H/3

A Steam Generator Tube Rupture (SGTR) has occurred and 2-OHP-4023-E-3, SGTR procedure, is being performed. The crew has isolated the ruptured Steam Generator and has completed the cooldown to a target temperature of <475.6°F.

The Unit Supervisor directs you to stabilize Reactor Coolant System (RCS) Temperature and set up Steam Dump Pressure Controller to maintain RCS temperature at approximately 470°F.

Which ONE of the following is the correct Steam Dump Pressure Controller setpoint required to maintain RCS temperature at approximately 470°F?

A. 446 psig

B. 500 psig

C. 530 psig

D. 543 psig

ANSWER: B

A. INCORRECT 446 psig is saturation temperature for 470 psig (460.67 -14.7 = 446).
B. CORRECT 514.67 psia is Psat for 470°F. The Controller would need to be set at 500 psig. (514.67 -14.7 = 500).
C. INCORRECT 530 psig is derived from adding 14.7 psi to 515 psia.
D. INCORRECT 543 psia is Psat for 475.6°F. (Absolute not PSIG)

OBJECTIVE: RO-C-05200-E9
REFERENCE: Steam Tables, SOD-05200-001

Attachment Provided : Steam Tables

KA - 000038 EK1.01
Steam Generator Tube Rupture (SGTR)
Knowledge of the operational implications of the following concepts as they apply to the SGTR:
Use of steam tables
RO - 3.1 SRO - 3.4
CFR - 41.8 / 41.10 / 45.3

KA Justification - Requires operator knowledge of how to use the steam tables to determine the required steam dump controller setpoint.

Original Question # - New 2012-048
Original Question KA - N/A

49. 049 004/RO/OK/NEW/NEW/000040 2.1.7/RO-C-EOP07-E4/RO-C-EOP07-E4 -1/H/3

Given the following plant conditions:

- Unit 1 Reactor Trip from 100% power.
- A Safety Injection has occurred due to Containment pressure.
- Pressurizer level and pressure are lowering rapidly.
- Reactor Coolant System temperature is 530°F and lowering.

Which ONE of the following is the cause of the above indications?

A. Steam Line Break

B. Steam Generator Tube Rupture

C. Pressurizer Safeties Stuck Open

D. Steam Generator Safeties Stuck Open

ANSWER: A

A. CORRECT	High Containment pressure, lowering temperature, pressure and level indicate a Steam Line Break.
B. INCORRECT	High Containment Pressure would not occur.
C. INCORRECT	Pressurizer Level would not be lowering.
D. INCORRECT	High Containment Pressure would not occur.

OBJECTIVE: RO-C-EOP07-E4
REFERENCE: RO-C-EOP07, Slide 23 & 27

KA - 000040 2.1.7
Steam Line Rupture
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 - Question requires operator to evaluate parameters and determine that a Steam Line rupture is occurring.

Original Question # - NEW 2012-049
Original Question KA - NA

50. 050 007/RO/OK/DIRECT/NRC EXAM 2004 -017-4/000056 AA1.07/RO-C-01900-E11/CM-39377/H/3

Unit 1 and Unit 2 were operating at 100% power with the Unit 1 and Unit 2 East Essential Service Water (ESW) pumps running with the Unit Crossties open.

Given the following sequence of events:

- Unit 2 tripped due to a turbine Electro-Hydraulic Control oil leak.
- Unit 1 remained on line.
- The Unit 2 Reserve Transformers are unavailable.
- Both Unit 2 Emergency Diesel Generators (EDGs) started.
- Buses T21A, T21B, and T21C were energized from the EDGs.
- Bus T21D failed to energize.

Assuming NO operator actions, which ONE of the following describes the ESW cooling water status for the Unit 2 EDGs?

A. 2CD EDG must be tripped immediately as ESW cooling has been lost.

B. 2CD EDG has ESW cooling supplied by the Unit 2 West ESW Pump.

C. 2AB EDG must be tripped immediately as ESW cooling has been lost.

D. 2CD EDG has ESW cooling supplied by the Unit 1 West ESW Pump.

ANSWER: D

A. INCORRECT	The Unit 1 West ESW pump will supply ESW Cooling water.
B. INCORRECT	The 2CD Diesel Generator could be supplied if the alternate ESW supply was manually opened (recent change).
C. INCORRECT	Diesel Generator 2AB has cooling from the auto start of the Unit 2 West ESW Pump (Would also have cooling from the Unit 1 East ESW).
D. CORRECT	When bus T21D is lost the Unit 2 East ESW Pump Trips, this will cause a low header pressure condition and automatically start the Unit 1 West ESW pump to Supply 2CD EDG with ESW cooling.

OBJECTIVE: RO-C-01900-E11
REFERENCE: RO-C-01900 TP-15, Unit 1 Essential Service Water
RO-C-01900 TP-16, Essential Service Water System Overview

KA - 000056 AA1.07
Loss of Offsite Power
Ability to operate and/or monitor the following as they apply to the Loss of Offsite Power:
Service water pump
RO - 3.2 SRO - 3.2
CFR - 41.7 / 45.5 / 45.6

KA Justification - Question requires operator to determine availability of the ESW pumps to supply desired load on a LOOP.

Original Question # - CM-39377, NRC EXAM 2004 -017-4, NRC EXAM 2002-095-1
Original Question KA - 000062 AA1.02

51. 051 003/RO/OK/DIRECT/NRC EXAM 2002-056-1/000057 AA2.18/RO-C-00300-E13/RO26-0144/H/3

Given the following plant conditions:

- The Unit was operating at 100% power when a Small Break LOCA has occurred.
- CRID 1 Instrument Bus tripped off just before the reactor trip.
- The RO turned both Safety Injection (SI) Actuation switches to ACTUATE on the Pressurizer Panel.

Which ONE of the following describes the status of the QMO-225, White SI Signal light and automatic/required manual actions for the QMO-225 East CCP Leak-off Valve?

The SI Signal light will ...

A. be lit since a manual SI has been performed. QMO-225 will cycle with pressure.

B. NOT be lit. The RO must manually actuate SI from the Safety Injection Panel switches to cause the light to illuminate and allow the valve to cycle with pressure.

C. NOT be lit. The RO must manually close QMO-225 if RCS pressure lowers below 1812 psig.

D. NOT be lit. QMO-225 will cycle with pressure since QMO-226 has the Safety Injection Signal light lit.

ANSWER: C

A. INCORRECT. SSPS output relays have lost power preventing the relay/light from energizing.

B . INCORRECT. SSPS output relays have lost power preventing the relay/light from energizing. The SI switches on the SIS panel do not change the status.

C. CORRECT. Since CRID 1 is lost, Train A SSPS output relays will not actuate. This is true regardless of which manual switch is turned. Since the relays do not energize, QMO-225 will not receive the SI signal and will not close on SI with low RCS pressure.

D. INCORRECT. The relay & associated functions are train Independent. Train A (QMO-225) will not operate with Train B (QMO-226)

OBJECTIVE: RO-C-0820360101-E1, RO-C-01101-E6, RO-C-00300-E13
REFERENCE: OHP-4023-E-0, Reactor Trip Or Safety Injection, pg 6-7;
OHP-4021-082-008, Operation of the CRID Power Supplies Table 1; RO-C-00300, pg 52

KA - 000057 AA2.18
Loss of Vital AC Electrical Instrument Bus
Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus:
The indicator, valve, breaker, or damper position which will occur on a loss of power
RO - 3.1 SRO - 3.1
CFR - 41.7 / 41.10 / 43.5 / 45.13

K/A Justification - Question tests ability to monitor the impact of the CRID loss on the SI signal to QMO-225, and the required manual actions based on this indication.

Original Question # - RO26-0144, RO 26 Audit-039, RO23 Audit-110-3 (RO# 090/SRO# N/A), NRC EXAM 2002-056-1
Original Question KA - 000057 AA1.06

52. 052 003/RO/OK/NEW/NEW/APE.058.AA1.02/RO-C-0820360101-E1/RO-C-08204-E4-1/H/3

Given the following plant conditions on Unit 2:

- The unit was operating at 100%.
- The Train 'A' 250 VDC, CRID 1 Inverter DC Input Breaker trips OPEN.

Which ONE of the following describes the status of CRID 1 and expected automatic actions, if any?

CRID 1 will be ...

A. energized from its respective 600 VAC alternate supply. The Auto Re-Transfer switch will attempt 1 transfer back to the normal supply if the DC Input breaker is manually reclosed.

B. energized from its respective 600 VAC alternate supply. The Auto Re-Transfer switch will attempt 1 reclosure of the DC Input breaker.

C. automatically re-energized from CRP-3. The Auto Re-Transfer switch will attempt 1 transfer back to the normal supply if the DC Input breaker is manually reclosed.

D. de-energized, but may be manually aligned to CRP-3. The Auto Re-Transfer switch will attempt 1 transfer back to the normal supply if the DC Input breaker is manually reclosed.

ANSWER: A

A. CORRECT. Auto transfer will occur to the vital bus on a loss of the normal 250 VDC feed to the inverter. 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.

B. INCORRECT. The static transfer switch provides a virtual zero time transfer to the alternate source in case of inverter failure. The transfer switch will NOT attempt to close the DC input breaker.

C. INCORRECT. The transfer is to 600 VAC vital bus. Plausible since CRP-3 provides backup power to the CRID Bus.

D. INCORRECT. The transfer is to 600 VAC vital bus. The transfer switch will NOT attempt to close the DC input breaker.

OBJECTIVE: RO-C-0820360101-E1, RO-C-08204-E4
REFERENCE: 2-OHP-4024-220, Drop 29; RO-C-AOP-D13

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

KA Justification - Requires the knowledge of the CRID Inverter (120 VAC Vital Power) response to a loss of 250 VDC and to the functioning of the inverter Auto Re-transfer operation.

Original Question # - NEW NRC EXAM 2012-052
Original Question KA - NA

53. 053 002/RO/OK/DIRECT/NRC EXAM 2004-015-2/000059 2.4.31/RO-C-02801B-E3/NRCAUDIT07-0011/F/4

Unit 1 has just entered mode 3 following a maintenance outage to replace a leaky fuel assembly. During work on the East RHR pump, an accidental spill causes radiation levels to increase.

The following radiation channels have alarmed:

- ERA-7305 U1 East RHR Pump Room - RED
- VRS-1505 Unit Vent Effluent Low Range Noble Gas - RED

Which ONE of the following is true regarding system operations based on these conditions?

A. The Auxiliary Building Supply fans have automatically tripped.

B. The AES Fan Charcoal Filter has automatically aligned.

C. The AES Fan Charcoal Filter must be manually placed in service.

D. The Auxiliary Building Supply fans must be manually tripped.

ANSWER: C

A. INCORRECT. Auxiliary Building Supply fans are not automatically tripped. (Fuel Pool area fans are tripped on local radiation)

B. INCORRECT. The dampers realign for flow through the charcoal filter bed when actuation by the manual selector switch or a Phase B actuation signal occurs.

C. CORRECT. If the alarm actuates on the ERA-7300 pump rooms the Operator is required to place the AES Fan Charcoal Filter test Selector Switch to the CHAR FILT position.

D. INCORRECT. Auxiliary Building Supply are not stopped.

OBJECTIVE: RO-C-02801B-E2, RO-C-02801B-E3
REFERENCE: 12-OHP-4024-139, Annunciator Response: Radiation Drop 11

KA - 000059 2.4.31
Accidental Liquid Radwaste Release
Emergency Procedures/Plan
Knowledge of annunciator alarms, indications, or response procedures.
RO - 4.2 SRO - 4.1
CFR - 41.10 / 45.3

KA Justification - Question requires operator knowledge of alarm response required for a radiation spill/release.

Original Question # - NRCAUDIT07-0011, NRC EXAM 2004-015-2
Original Question KA - 000059 AK2.02

54. 054 003/RO/OK/DIRECT/NRC EXAM 2007-60/000061 AA1.01/RO-C-02801A-E8/2007-0524/F/3

Which ONE of the following describes the Control Room Ventilation System pressurization fan alignment following receipt of a ERS 8401 Control Room Radiation Monitor High alarm?

A. Both Unit 1 Control Room Pressurization Fans are RUNNING
Both Unit 2 Control Room Pressurization Fans are RUNNING

B. Both Unit 1 Control Room Pressurization Fans are STOPPED
Both Unit 2 Control Room Pressurization Fans are RUNNING

C. Both Units West Control Room Pressurization Fans are RUNNING
Both Units East Control Room Pressurization Fans are STOPPED

D. Both Units West Control Room Pressurization Fans are STOPPED
Both Units East Control Room Pressurization Fans are RUNNING

ANSWER: B

A. INCORRECT. Only the Unit 2 Fans Operate off of ERS-8401
B. CORRECT. Each Control Room has a Separate Control Room Ventilation System. Each Control Room Has a Separate Radiation Monitor which controls its associated fans. ERS-7400 for Unit 1 and ERS-8400 for Unit 2.
C. INCORRECT. Both Unit 2 Fans Operate off of ERS-8401 (True for a train of SI)
D. INCORRECT. Both Unit 2 Fans Operate off of ERS-8401 (True for a train of SI)

OBJECTIVE: RO-C-02801A-E8
REFERENCE: SOD-01350-001, SOD-02801A-001, RO-C-02801A pg. 20
KA - 000061 AA1.01
Area Radiation Monitoring (ARM) System Alarms
Ability to operate and/or monitor the following as they apply to the Area Radiation Monitoring (ARM) System Alarms:
Automatic actuation
RO - 3.6 SRO - 3.6
CFR - 41.7 / 45.5 / 45.6

KA Justification - Question requires operator to determine which pressurization fans are automatically started due to a control room area radiation monitor.

Original Question # - NRCAUDIT07-0950,2007-0524, NRC Exam 2007-60
Original Question KA - 072000 K1.04

55. 055 002/RO/OK/DIRECT/NRC EXAM 2006-020-4/000062 AK3.03/RO-C-AOP0590412-E3/CM-7839/F/3

Unit 1 is operating at 100%, steady state condition, when Annunciator Panel 118, Drop 84, ESW PIPE TUNNEL SUMP LEVEL HI-HI, alarm is received. East ESW supply and return flows indicate an abnormal high differential.

Unit 2 has reported that 2-OHP-4022-019-001, ESW SYSTEM LOSS/RUPTURE has been implemented.

What action(s) would be taken by BOTH Unit control room operators?

A. Close affected header unit crosstie valves.

B. Align alternate ESW cooling to non affected diesel generators.

C. Stop affected header ESW pumps.

D. Place associated diesel generators in tripped condition.

ANSWER: A

A. CORRECT	When ESW supply and return flows indicate an abnormal high differential both units are directed to isolate the affected header unit crosstie valves.
B. INCORRECT	The alternate ESW cooling to the EDGs is isolated per step 15.
C. INCORRECT	The header crosstie valves are closed first to prevent loss of the opposite unit and then only the affected unit will need to stop pumps.
D. INCORRECT	The affected EDG is placed in the tripped condition after the headers are isolated and ESW pumps are stopped.

OBJECTIVE: RO-C-AOP0590412-E2, RO-C-AOP0590412-E3
REFERENCE: 1-OHP 4022.019.001, ESW SYSTEM RUPTURE, pg 4-5

KA - 000062 AK3.03
Loss of Nuclear Service Water
Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water:
Guidance actions contained in EOP for Loss of nuclear service water
RO - 4.0 SRO - 4.2
CFR - 41.5 / 41.10 / 45.6 / 45.13

KA Justification - Question requires operator knowledge of the major action contained in the Loss of ESW EOP(AOP).

Original Question # - Cook NRC Exam 2006-020-4 : Bank 12AOPS0515-1, NRCAUDIT07-0730, CM-7839
Original Question KA - APE.062.AK3.03

56. 056 005/RO/OK/DIRECT/NRC EXAM 2007-67/000065 2.1.32/RO-C-EC01-E4/2007-0546/H/3

Unit 2 is performing OHP-4022-064-002 Loss of Control Air Recovery procedure. All RCPs have been tripped. You are told to initiate a cooldown.

Which ONE of the following describes the method used to perform a RCS cooldown and the concerns?

Nitrogen must be locally aligned to the Steam Generator (SG) PORVs and then the cooldown is performed by...

A. evenly steaming all 4 SGs from the Control Room SG PORV Controllers to prevent uneven cooling which could lead to a SI.

B. steaming SGs #1 & 2 from the Control Room SG PORV Controllers to prevent excessive cooldown in the Pressurizer loop which could lead to loss of level.

C. directing operators stationed at #1/4 & #2/3 SG PORV Emergency Control Loader valves to evenly steam all 4 SGs to prevent uneven cooling which could lead to a SI.

D. directing an operator to steam SGs #1 & 4 from the SG PORV Emergency Control Loader valves to prevent excessive cooldown in the Pressurizer loop which could lead to loss of level.

ANSWER: C

A. INCORRECT When on Backup Nitrogen, the Local Control Stations must be used.

B. INCORRECT When on Backup Nitrogen, the Local Control Stations must be used. The Cooldown rate is limited to prevent loss of PRZ level but cooling must be even.

C. CORRECT The Loss of Air Causes Steam Stops to Fail Closed. After Nitrogen is aligned to the SG PORVs, They must be controlled locally using the SG PORV Emergency Control Loader valves (Two Stations each with 2 controllers for #21/24 SG PORVs & #22/23 SG PORVs). With the RCPs Stopped the cooldown must be evenly performed to prevent lowering 1 SG pressure below the others. If 1 Pressure is < 100 psig below 2 others an SI signal is generated.

D. INCORRECT The Cooldown rate is limited to prevent loss of PRZ level but cooling must be even.

OBJECTIVE: RO-C-AOP0490412-E3, RO-C-EC01-E4
REFERENCE: 2-OHP-4022-064-002, pgs 7 & 38.

KA - 000065 2.1.32
Loss of Instrument Air
Conduct of Operations
Ability to explain and apply system limits and precautions.
RO - 3.8 SRO - 4.0
CFR - 41.10 / 43.2 / 45.12

KA Justification - Question Requires operator to describe the method used to address the cooldown following a loss of IA and the associated precautions with this method.

Original Question # - 2007-0546, NRC EXAM 2007-67
Original Question KA - Generic 2.1.30

57. 057 005/RO/OK/DIRECT/NRC EXAM 2006-023-6/000076 2.1.30/RO-C-0030500401-E2/NRCAUDIT07-0764/F/4

Given the following conditions on Unit 2:

- A reactor trip occurred from 100% power.
- Chemical and Volume Control System (CVCS) Letdown is at 75 gpm.
- The operators have transitioned from OHP-4023-E-0 Reactor Trip or Safety Injection to OHP-4023-ES-0.1 Reactor Trip Response.
- ERA-8309, Reactor Coolant Filter Room has a high alarm.
- ERA-8303, East CCP Room shows a rising trend, but has yet to alarm.
- The SRO implements 12-OHP-4022-002-019, High Reactor Coolant Activity or Failed Fuel.

Which ONE of the following describes the correct actions to take with the CVCS and the basis for these actions?

A. Charging and Letdown are isolated to contain the high radioactivity within the containment building.

B. Letdown is diverted to the CVCS HUT to limit radiation levels in the charging pump area.

C. Charging and Letdown are maximized through the letdown demineralizers to maximize clean up.

D. Excess Letdown is placed in service through the letdown demineralizers to maximize clean up.

ANSWER: C

A - INCORRECT. CVCS is used to cleanup the RCS so flow is maximized not isolated.

B - INCORRECT. Letdown is not diverted. The flow is maximized through the demineralizers to cleanup the RCS. Diverting would create added waste.

C - CORRECT. 12-OHP-4022-002-019 directs the operator to verify Letdown lineup and maximize letdown flow to help reduce RCS Activity.

D - INCORRECT. Excess letdown does not flow through the letdown demineralizers.

OBJECTIVE: RO-C-0030500401-E2, RO-C-0030500401-E3
REFERENCE: 12-OHP-4022-002-019, High Reactor Coolant Activity or Failed Fuel

KA - 000076 2.1.30
High Reactor Coolant Activity
Conduct of Operations
Ability to locate and operate components, including local controls.
RO - 4.4 SRO - 4.0
CFR - 41.7 / 45.7

KA Justification - Question requires operator to determine which controls are operated to address the high radioactivity in the RCS and CVCS.

Original Question # - NRCAUDIT07-0764, CM-7873, RO27 Audit-24, Cook NRC Exam 2006-023-6 : INPO # 26703 Indian Point 3 (Unit) - 12/11/2003
Original Question KA - 000076 AA2.02, 000076 AK3.05

58. 058 007/RO/OK/NEW/NEW/000077 AK1.03/RO-C-0821040401-E1/RO-C-0821040401-E1-1/H/3

Given the following plant conditions on Unit 1:

- Plant is at 100% power.
- Generator output: 1100 MWe.
- Generator Reactive MVA Indication: 50 MVAR OUT.
- All systems normally aligned.
- The Transmission Operator has notified the plant that system grid voltage is high and forecast to go higher.
- The Transmission Operator requests the plant to take actions to help stabilize the grid.

Which ONE of the following describes the changes that will be required by the operator and the expected indications?

The exciter field current will need to be ...

A. REDUCED causing the Main Generator to become OVER-Excited as indicated by Negative MVARs (VARs IN).

B. REDUCED causing the Main Generator to become UNDER-Excited as indicated by Negative MVARs (VARs IN).

C. RAISED causing the Main Generator to become OVER-Excited as indicated by Positive MVARs (VARs OUT).

D. RAISED causing the Main Generator to become UNDER-Excited as indicated by Positive MVARs (VARs OUT).

ANSWER: B

A. INCORRECT	Exciter field current will be lowered but this leads to Negative MVARS and under-excited conditions.
B. CORRECT	Exciter field current will be lowered to lower voltage leading to Negative MVARS and under-excited conditions.
C. INCORRECT	Exciter field current will be lowered and this leads to Negative MVARS and under-excited conditions.
D. INCORRECT	Exciter field current will be lowered and this leads to Negative MVARS and under-excited conditions.

OBJECTIVE: RO-C-0821040401-E1; RO-C-GF05-E20; RO-C-08001-E8
REFERENCE: RO-C-GF05

KA - 000077 AK1.03
Generator Voltage and Electric Grid Disturbances
Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances:
Under-excitation
RO - 3.3 SRO - 3.4
CFR - 41.4 / 41.5 / 41.7 / 41.10 / 45.8

KA Justification - Question requires candidate to determine correct direction to adjust current and understand the impact to the Main Generator and expected indications.

Higher Order Justification - Comprehensive knowledge requires operator to recognize relationships and interactions between voltage, excitation, and MVARs.

Original Question # NEW NRC Exam 2012-058
Original Question KA - NA

59. 059 008/RO/OK/DIRECT/RO25 AUDIT-37/00WE03 EK1.2/RO-C-EOP09-E25/NRCAUDIT07-1001/H/3

Following a SBLOCA, you have entered the ECCS flow reduction sequence in OHP-4023-ES-1.2 Post LOCA Cooldown and Depressurization.

You are preparing to stop the first CCP. RCS subcooling exactly meets the subcooling table requirements to do this.

After you stop the pump, you expect subcooling to:

A. Lower more than it will when you isolate the BIT later.

B. Lower then return to the original value when pressure stabilizes.

C. Lower to the minimum required value then recover with the cooldown.

D. Lower to the point at which SI must be reinitiated.

ANSWER: C

A. INCORRECT Isolating the BIT will cause a greater flow reduction than stopping the single pump.

B. INCORRECT Subcooling will not return to previous value when pressure stabilizes.

C. CORRECT With break flow now greater than ECCS flow, RCS pressure begins to decrease as inventory decreases and the bubble in the PRZ expands. As RCS pressure decreases, ECCS flow increases and break flow decreases. Eventually a new equilibrium point is reached where the ECCS flow from 1 CCP + SI pumps is equal to the reduced break flow at the new lower equilibrium RCS pressure. The RCS will remain subcooled at this new lower equilibrium pressure.

D. INCORRECT The Required Subcooling values are designed to ensure that the reduction will not be great enough to require SI re-initiation.

OBJECTIVE: RO-C-EOP09-E25, RO-C-EOP02-E17
REFERENCE: OHP-4023-ES-1.2 Post LOCA Cooldown and Depressurization, pg 16.

KA - 00WE03 EK1.2
LOCA Cooldown and Depressurization
Knowledge of the operational implications of the following concepts as they apply to the LOCA Cooldown and Depressurization:
Normal, abnormal and emergency operating procedures associated with LOCA Cooldown and Depressurization
RO - 3.6 SRO - 4.1
CFR - 41.8 / 41.10 / 45.3

KA Justification - Question requires operator knowledge of the implication that stopping pumps per the procedure has on the RCS Subcooling.

Original Question # - NRCAUDIT07-1001, MASTER 01EOPC0217-1, RO25 Audit-3
Original Question KA - 000009 EK3.26 , EPE E03 EK3.1

60. 060 002/RO/OK/DIRECT/RO27 AUDIT-16/00WE04 EK2.2/RO-C-EOP09-E34/RO-C-EOP09-E34-1/H/3

Given the following conditions:

- A LOCA outside containment has resulted in RCS subcooling dropping to 0°F.
- The operating crew has entered OHP-4023-ECA-1.2, LOCA Outside Containment.

Which ONE of the following identifies the expected status of Containment Phase A Isolation, and the parameter used to verify that the LOCA has been isolated in accordance with ECA-1.2?

	Phase A Status	Parameter Monitored for LOCA Isolation
A.	NOT automatically actuated	Pressurizer level rising
B.	NOT automatically actuated	RCS pressure rising
C.	automatically actuated	Pressurizer level rising
D.	automatically actuated	RCS pressure rising

ANSWER: D

A - INCORRECT. Pressurizer level rising is plausible since the student could reason that it may be rising if the leak was isolated. The procedure directs the use of RCS pressure increasing as the method used to indicate the leak has been isolated. Phase A not being actuated is plausible if applicant misapplies the fact that the LOCA was outside of containment, and therefore containment pressure does NOT rise to 1.0 psig. 1.0 psig is the containment pressure setpoint for Safety Injection. Since Phase A isolates certain components of containment, and since this LOCA is outside containment, it is plausible that the applicant would believe that a Phase A is not needed.

B - INCORRECT. Plausible since the procedure does direct the use of RCS pressure increasing as the method for determining the leak is isolated.

C - INCORRECT. Plausible since a Phase A has been actuated, due to the Safety Injection signal. Pressurizer level rising is plausible since the student could reason that it may be rising if the leak was isolated.

D - CORRECT. The procedure directs the use of RCS pressure increasing as the method used to indicate the leak has been isolated. A Phase A has actuated, since a Safety Injection was automatically initiated due to low RCS pressure resulting from the LOCA.

OBJECTIVE: RO-C-EOP09-E34
REFERENCE: OHP-4023-E-0, pg 22; OHP-4023-ECA-1.2, pg 4

KA - 00WE04 EK2.2
LOCA Outside Containment
Knowledge of the interrelations between the LOCA Outside Containment 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 - 3.8 SRO - 4.0
CFR - 41.7 / 45.7

KA Justification - Requires knowledge if the automatic feature actuations (SI/Phase A Actuation) that will occur and the manual means of verifying LOCA isolations and proper operation of systems (RCS Pressure rising) used during a LOCA outside of containment event.

Original Question #- RO-C-EOP09-E34-1, RO27 Audit-16
Original Question KA - 00WE04 EK2.2

61. 061 002/RO/OK/DIRECT/NRC EXAM 2004-027/00WE06 EK3.1/RO-C-EOP10-E12/CM-39319/H/3

The crew has entered OHP-4023-FR-C-2, Response to Degraded Core Cooling.

The following conditions exist:
- RCS Hot Leg Temperatures are 300°F.
- Reactor Vessel Level Indication System (RVLIS) NR indications are 37%.

Which ONE of the following would be most effective in restoring core cooling?

A. Depressurizing SGs to Atmospheric Pressure.

B. Aligning BIT flow from the Opposite Unit.

C. Starting a Residual Heat Removal Pump.

D. Starting a Reactor Coolant Pump.

ANSWER: C

A. INCORRECT	Further depressurization of the SGs will not significantly add to RCS cooling. With RCS hot leg temperatures at 300°F SG pressures would be ~52 psig. The RCS needs inventory makeup to restore cooling.
B. INCORRECT	BIT flow from the opposite unit will be limited to ~ 50 gpm.
C. CORRECT	An RHR pump will provide the greatest injection flow to restore inventory and thus will restore core cooling. At this temperature with the RCS Saturated, pressure will be less than 60 psig resulting in ~3000 gpm from the RHR pump.
D. INCORRECT	Without makeup to the RCS starting the RCPs will not significantly add to core cooling.

OBJECTIVE: RO-C-EOP10-E12
REFERENCE: 12-OHP-4023-FR-C-2, Response to Degraded Core Cooling Background Document Step 19 and 20 Basis pg. 37 and 40, RO-C-EOP10 pg. 66-68, and 77-78

KA - 00WE06 EK3.1
Degraded Core Cooling
Knowledge of the reasons for the following responses as they apply to the Degraded Core Cooling:
Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics
RO - 3.4 SRO - 3.8
CFR - 41.5 / 41.10 / 45.6 / 45.13

KA Justification - Question requires operator knowledge of the RHR pump characteristics (flow/pressure) compared to the distractors and the RCS characteristics (temp/press) in order to determine the best method of cooling.

Original Question # - CM-39319, NRCAUDIT07-0170, NRC EXAM 2004-027, Master Bank EOP15~47, NRC Exam 2002-021-002, 22817-COOK02
Original Question KA - 00WE06 EK1.1

62. 062 003/RO/OK/DIRECT/RO22 AUDIT- BOTH-78/00WE08 EK2.1/RO-C-EOP12-E28/CM-39624/H/3

Given the following events and conditions:

- Unit 1 is responding to a steam break inside containment while at full power.
- All systems operate as designed.
- Narrow range Steam Generator (SG) level is 15% for each intact SG.
- AFW flow is 40 X 10^3 lbm/hr to each intact SG.
- The Reactor Coolant Pumps (RCP) were tripped.

The crew entered OHP-4023-FR-P-1, Response to Imminent Pressurized Thermal Shock Condition, due to low temperature:

- RCS temperature is now stable.
- RCS pressure is stable with only the control group of pressurizer heaters energized.
- Letdown has been restored .
- The crew has determined that a one-hour soak is required.

Which ONE of the following evolutions could be performed by the crew in the next hour while continuing through the procedures?

A. Start #13 RCP.

B. Place auxiliary spray in service.

C. Raise AFW flow to one intact SG to raise NR level to >25%.

D. Commence a 25°F/hour cooldown to Mode 5.

ANSWER: B

A. INCORRECT Starting a RCP will cause a pressure transient and could cause further cooldown.
B. CORRECT Any actions that will NOT cause either a cooldown or a pressure rise and are specified by any other procedure in effect are permitted during this soak period.
C. INCORRECT Increases cooldown stressing the vessel.
D. INCORRECT Cooldown is NOT allowed.

OBJECTIVE: RO-C-EOP12-E28
REFERENCE: 1-OHP-4023-FR-P-1, Response to Imminent Pressurized Thermal Shock Condition, step 26

KA - 00WE08 EK2.1
Pressurized Thermal Shock
Knowledge of the interrelations between the Pressurized Thermal Shock and the following:
Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features
RO - 3.4 SRO - 3.7
CFR - 41.7 / 45.7

KA Justification - Questions requires candidate knowledge of the components listed within the distractors and their functions/interactions with the RCS and impact on the associated cooldown restriction.

Original Question # - RO22 AUDIT- BOTH-78, CM-39624
Original Question KA - WE 08 EK 3.3

63. 063 006/RO/OK/NEW/NEW/00WE11 2.1.32/RO-C-EOP09-E36/RO-C-EOP09-E36-2/F/3

Given the following conditions on Unit 2:

- The plant was operating steady-state at 100% power.
- A Reactor Trip and Safety Injection have occurred due to a LOCA outside containment.
- The crew has performed all the applicable procedure steps of OHP-4023-ECA-1.2, LOCA Outside Containment.
- The LOCA has **NOT** been isolated, and OHP-4023-ECA-1.1, Loss of Emergency Coolant Recirculation, has been implemented.

Which ONE of the following states the basis for cooling the Reactor Coolant System (RCS) at a maximum cooldown rate during the implementation of OHP-4023-ECA-1.1?

The cooldown rate will...

A. prevent void formation in the head.

B. minimize offsite releases due to this event.

C. ensure adequate shutdown margin is maintained.

D. limit the depletion of the Condensate Storage Tank (CST).

ANSWER: B

A. INCORRECT. At a 100°F/hr rate, voids may still occur in the RCS, however this is not a concern since a LOCA already exists.

B. CORRECT. By reducing the overall temperature of the RCS (including metal), a reduction of the need for supporting systems and equipment needed to remove heat is obtained. This action will minimize any offsite releases during the event.

C. INCORRECT. The cooldown rate does not ensure that SDM is maintained. Chem. samples are required to ensure adequate SDM. SDM should be maintained since ECCS flow was established due to the LOCA.

D. INCORRECT. A LOCA has occurred. The primary concern is RCS leakage outside of containment and the resultant release dose. Limiting CST depletion is a concern during natural circulation and to a lessor extent during ECA-1.1 but is not the reason for the maximum rate cooldown.

OBJECTIVE: RO-C-EOP09-E36
REFERENCE: 12-OHP-4023-ECA-1.1, Step 9 Background.

KA - 00WE11 2.1.32
Loss of Emergency Coolant Recirculation
Conduct of Operations
Ability to explain and apply system limits and precautions.
RO - 3.8 SRO - 4.0
CFR - 41.10 / 43.2 / 45.12

KA Justification - Requires the ability to explain the basis for cooldown rate limits imposed in ECA-1.1, Loss of Emergency Coolant Recirculation.

Original Question # - NRC Exam 2012-063
Original Question KA - NA

64. 064 006/RO/OK/DIRECT/RO23 AUDIT-003-1/00WE14 EA2.2/RO-C-EOP09-E42/NRCAUDIT07-0388/F/3

Unit 1 experienced a large break LOCA.

The following conditions exist:

- E RHR Pump is tagged out for maintenance.
- Containment pressure is stable at 9 psig.
- RWST level has lowered to less than 25%.

The crew has performed OHP-4023-FR-Z.1, Response to High Containment Pressure. The crew implemented OHP-4023-ECA-1.1, Loss of Emergency Coolant Recirculation, after the Recirc Sump to West RHR/CTS Pumps' valve (ICM-306) failed CLOSED while attempting to open the valve per OHP-4023-ES-1.3, Transfer to Cold Leg Recirculation.

ASSUME: E CTS Pump running on the Recirc. Sump.

NOTE: OHP-4023-FR-Z.1 requires both CTS pumps to be in operation, but OHP-4023-ECA-1.1 limits the operators to only one CTS pump.

Which ONE of these two procedures takes priority under these conditions and what is the basis for this requirement?

A. OHP-4023-FR-Z.1 takes priority because a total loss of RHR causes the CTS system to become relatively more important to reduce containment pressure.

B. OHP-4023-FR-Z.1 takes priority because red path FRPs always have priority over ECA procedures.

C. OHP-4023-ECA-1.1 takes priority because it conserves RWST water level as long as possible for injection while providing sufficient CTS flow to mitigate containment pressure.

D. OHP-4023-ECA-1.1 takes priority because ECA procedures always have priority over FRPs.

ANSWER: C

A. INCORRECT ECA-1 .1 CTS operation takes priority over FR-Z.1 Plausible: Although a loss of RHR and containment sump recirc causes a loss of the containment heat sink, the supply for SI comes from the RWST which will be drawn down until containment sump recirculation can be established.

B. INCORRECT ECA-1 .1 CTS operation takes priority over FR-Z.1 Plausible: FRPs normally take priority over most EOPs.

C. CORRECT ECA-1 .1 operation of the CTS pumps takes priority over FR-Z.1 in order to limit RWST usage.

D. INCORRECT ECAs do not always have priority over FRPs. Plausible: Some ECAs take priority e.g. ECA-0.0 has priority over FRPs in that FRPs are suspended until power is restored.

OBJECTIVE: RO-C-EOP09-E42
REFERENCE: 1-OHP-4023-ECA-1.1 Loss of Emergency Coolant Recirculation, Step 5 , 12-OHP-4023-FR-Z.1, Background Document pg. 5 Step 2 Basis

KA - 00WE14 EA2.2
High Containment Pressure
Ability to determine and interpret the following as they apply to the High Containment Pressure:
Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments
RO - 3.3 SRO - 3.8
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Question requires operator ability to determine the correct procedural priority based on limitations (loss of recirc/RWST level) and precautions (Z.1) and the reason.

Original Question # - NRCAUDIT07-0388, RO23 AUDIT-003-1
Original Question KA - 2.4.23

65. 065 002/RO/OK/DIRECT/RO27 AUDIT-27/00WE15 EK3.2/RO-C-EOP13-E3/NRCAUDIT07-0644/F/3

Given the following conditions on Unit 1:
- The Unit has experienced a rupture of the NESW piping inside containment.

The following conditions now exist:
- NLI-330/331 MIN RECIRC LEVEL lights are LIT.
- NLI-340/341 FLOOD LEVEL lights are LIT.

Why is safe plant recovery not assured for a design basis Large Break LOCA when FLOOD LEVEL lights are LIT?

A. Operation of critical ECCS components needed for safe recovery is endangered by submersion.

B. Operation of the CTS pumps is endangered by excess debris fouling the containment suction strainers.

C. Operation of the hydrogen recombiners is compromised by loss of direct access to the containment atmosphere.

D. Operation of the RHR system is compromised by high suction pressure

ANSWER: A

A - CORRECT. Containment design basis flood level takes into account the entire water contents of the RCS, RWST, Ice condenser ice bed melt, and SI accumulators, plus the added mass of a LOCA and a steam line or feedline break inside containment. NESW and CCW may be major contributors to exceeding "flood" level and causing a loss of equipment required for long term cooling.

B - INCORRECT. The Containment Sump strainer debris fouling is not significantly impacted by the higher level.

C - INCORRECT. The hydrogen recombiner elevation is above the flood stage and is not impacted due to its higher elevation.

D - INCORRECT. The RHR Suction pressure may be slightly higher but the pumps are designed for use with a pressurized RCS during Shutdown operations.

OBJECTIVE: RO-C-EOP13-E3
REFERENCE: 12-OHP-4023-FR-Z-2, Response to Containment Flooding Background pg. 4 (Step 1)

KA - 00WE15 EK3.2
Containment Flooding
Knowledge of the reasons for the following responses as they apply to the Containment Flooding:
Normal, abnormal and emergency operating procedures associated with Containment Flooding
RO - 2.8 SRO - 3.1
CFR - 41.5 / 41.10 / 45.6 / 45.13

KA Justification - Requires knowledge of the basis behind EOP steps associated with containment flooding.

Original Question # - NRCAUDIT07-0644, RO27 Audit-27, AUDIT RO23-088-7
Original Question KA - 00WE15 EK1.2, W/E 15 G2.1.7

66. 066 005/RO/OK/DIRECT/WOLF CREEK 2011/194001 2.1.4/RO-C-ADM14-E18/RO-C-ADM14-E18-1/F/3

Given the following plant conditions:

- All shifts are manned to the minimum composition per Technical Specifications.
- One RO becomes seriously ill and must be taken to the hospital.
- There are four hours left until shift change.

Which ONE of the following describes the required actions?

A. Action must be taken within one hour to identify a relief operator who will arrive within the following three hours.

B. The affected operator must not be allowed to leave site until a relief operator arrives.

C. Action must be taken to ensure a relief operator arrives within two hours.

D. No action is required since turnover will occur within four hours.

ANSWER: C

A. INCORRECT	The operator must be replaced within 2 hours. Plausible since typical TS required actions are required within 1 hour to start and/or 4 hours total.
B. INCORRECT	The Operator must leave (FFD requirements apply), TS Allows for unexpected absences.
C. CORRECT	TS requires that action be taken to find a replacement. The position cannot remain unmanned for greater than 2 hours.
D. INCORRECT	Incorrect, must be filled in 2 hours (not the typical TS 4 hour allowance)

OBJECTIVE: RO-C-ADM14-E18
REFERENCE: Technical Specification 5.2.2

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 crew responsibility for maintaining minimum staffing in the event of illness, unexpected absence.

Original Question # - WOLF CREEK 2011
Original Question KA - 194001 2.1.4

67. 067 008/RO/OK - ATTACHMENT/DIRECT/RO22 AUDIT- RO1/194001 2.1.5/RO-C-ADM14-E18/NRCAUDIT07-0651/H/3

Given the following plant conditions:

- Unit 1 is shutdown with the RCS at 2085 psig and 547ºF.
- Unit 2 is shutdown with the RCS at 325 psig and 170ºF.

In addition to the Shift Manager, Unit Supervisors, STA, and WCC-SRO; minimum shift staffing for the station also requires which ONE of the following under the conditions shown above?

Attachment Provided

	Reactor Operators	Qualified Operators
A.	3	8
B.	4	3
C.	3	4
D.	4	8

ANSWER: D

A. INCORRECT 4 ROs are required, 2 for Unit 1, 1 for Unit 2, and a shared RO

B. INCORRECT 8 QOs are required, 2 for Unit 1, 1 for Unit 2, and 4 shared QOs

C. INCORRECT 4 ROs & 8 QOs are required, 2/2 for Unit 1, 1/1 for Unit 2, and 1/4 shared

D. CORRECT The Conditions place Unit 1 in Mode 3 and Unit 2 In Mode 5. This requires staffing per the second table of Section 3.6 In OHI-4000 Attachment 22.

OBJECTIVE: RO-C-ADM14-E18
REFERENCE: OHI-4000, Conduct of Operations: Standards Attachment 22

Attachment Provided: OHI-4000, Conduct of Operations: Standards Attachment 22

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 operator ability to determine correct mode and the associated shift manning requirement per the procedure.

Original Question # - NRCAUDIT07-0651, Audit RO22 - RO1, NRCAUDIT07-0322
Original Question KA - P2.1.4

68. 068 006/RO/OK/DIRECT/NRC EXAM 2004-009-1/194001 2.1.36/RO-C-ADM13-E3/NRCAUDIT07-0007/F/3

Core Alterations are in progress on Unit 2. RCS boron concentration has been verified to be 2360 ppm (two samples analyzed).

The crew is required to ...

A. suspend core alterations and positive reactivity changes, and initiate boration.

B. suspend core alterations and positive reactivity changes, and establish containment integrity.

C. suspend core alterations and remove all personnel from the containment building.

D. remove all personnel from the containment building, establish containment integrity, and initiate boration.

ANSWER: A

A. CORRECT. Technical Specification 3.9.1 requires either Keff <COLR limit. 2-OHP-4030-227-037 requires the most conservative of the Keff or 2400 ppm.
B. INCORRECT. Containment Integrity is not required.
C. INCORRECT. Containment evacuation is not required.
D. INCORRECT. Containment evacuation and Containment Integrity are not required.
OBJECTIVE: RO-C-ADM13-E3, RO-C-ADM13-E8
REFERENCE: Technical Specification 3.9.1 Refueling Boron Concentration

KA - 194001 2.1.36
Generic
Conduct of Operations
Knowledge of procedures and limitations involved in core alterations.
RO - 3.0 SRO - 4.1
CFR - 41.10 / 43.6 / 45.7

KA Justification - Question tests Knowledge of Technical Specification Limit and administrative requirement to Stop Core Alts when Boron is < 2400 ppm.

Original Question # - NRCAUDIT07-0007, RO26 AUDIT-071, NRC Exam 2004-009-1, 19249 - COOK01
Original Question KA - 194001 2.1.40, APE.036.AK1.02

69. 069 012/RO/OK/DIRECT/RO27 AUDIT-18/194001 2.2.36/RO-C-05103-E11/RO27AUDIT-18/H/3

Given the following conditions on Unit 2:

- Unit is in Mode 1 at 10%
- In order to repair broken conduit, power must be removed from the air supply solenoid for MRV-211, SG Stop Valve 1 (MRV-210) Dump Valve.

Which ONE of the following identifies the:

1) response of the MRV-210, Steam Generator Stop Valve when power is procedurally removed from the Dump Valve, MRV-211 and,
2) required Technical Specification LCO entry, if any?

A. 1) MRV-210 will fail CLOSED
2) Technical Specification entry is required.

B. 1) MRV-210 will fail CLOSED
2) Technical Specification entry is NOT required.

C. 1) MRV-210 will remain OPEN
2) Technical Specification entry is required.

D. 1) MRV-210 will remain OPEN
2) Technical Specification entry is NOT required.

ANSWER: C

A - INCORRECT. The Stop Valve will remain open. Tech Spec entry (3.3.2 and 3.7.2) is required.
B - INCORRECT. The Stop Valve will remain open. Tech Spec entry (3.3.2 and 3.7.2) is required.
C - CORRECT. The Stop Valve will remain open. Removing power from the dump valve solenoid prevents it from opening to dump steam. Tech Spec entry is required since one train of isolation is disabled (3.3.2 and 3.7.2).
D - INCORRECT. The Stop Valve will remain open. Tech Spec entry is required even though the valve will still close. One train of isolation is disabled (3.3.2 and 3.7.2).

OBJECTIVE: RO-C-05103-E4, RO-C-05103-E5, RO-C-05103-E11
REFERENCE: RO-C-05103 pg. 21, TS 3.3.2 and 3.7.2

KA - 194001 2.2.36
Generic
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 operator knowledge of the failure direction of the Steam Stop Dump valve due to a loss of power and the impact of this event on the operability status.

Original Question # - RO27AUDIT-18, WattsBarMay2009-18
Original Question KA - 00WE12 2.2.36

70. 070 003/RO/OK/DIRECT/NRC EXAM 2006-022-10/194001 2.2.42/RO-C-03400-E12/CM-40016/F/3

Which ONE of the following conditions would require an LCO Action Statement entry for Technical Specification Section 3.6, Containment Systems?

A. While in MODE 1, an electrician opens the outer airlock door at the lower containment access without prior approval.

B. While in MODE 3, during an inspection of an containment equipment hatch, it is determined that the equipment hatch is NOT sealed.

C. While in MODE 4, Containment internal pressure is found to be -0.5 psig prior to placing Containment Purge in service.

D. While in MODE 5, during performance of the Overall Integrated Containment Leakage Rate Test, Containment leakage exceeds the maximum allowable Technical Specification leakage rates.

ANSWER: B

A. INCORRECT	One airlock door may be opened for normal entry / exit while in Modes 1-4 per T.S. 3.6.2.
B. CORRECT	The airlock door interlocks are required to be operable in Modes 1 - 4 per T.S. 3.6.2.
C. INCORRECT	Containment pressure limits are -1.5 to +0.3 psig during Modes 1-4 per T.S. 3.6.4.
D. INCORRECT	Overall Containment leak rate only applies in Modes 1-4 per T.S. 3.6.1

OBJECTIVE: RO-C-03400-E12
REFERENCE: Tech Spec 3.6.2

KA - 194001 2.2.42
Generic
Equipment Control
Ability to recognize system parameters that are entry-level conditions for Technical Specifications.
RO - 3.9 SRO - 4.6
CFR - 41.7 / 41.10 / 43.2 / 43.3 / 45.3

KA Justification - Question requires operator to determine which of the conditions would require entry into the Containment TS.

Original Question # - CM-40016, CM-7824, NRCAUDIT07-0174, RO25 AUDIT-24, Cook NRC Exam 2006-022-10, INPO # 22812 Cook 1 - 12/9/2002 #016 (RO#12/SRO#15)
Original Question KA - 000069 AK2.03

71. 071 005/RO/OK/DIRECT/RO25 AUDIT-73/194001 2.3.13/RO-C-0280040501-E3/CM-1132/F/2

When changing the configuration of the Aux Building Ventilation System, ______________ should be notified to ensure ________________________

A. the Shift Manager;
that management personnel are aware of current Aux Building Ventilation configuration in the event of an accident.

B. Maintenance;
that the fans are checked for proper operation prior to running for prolonged periods of time.

C. Radiation Protection;
that the radiological conditions in the Aux Building are appropriately monitored.

D. Work Control;
that the surveillance requirements for the ventilation systems are up-to-date and remain current.

ANSWER: C

A. INCORRECT	Wrong person notified. Incorrect reason for notification.
B. INCORRECT	Wrong person notified. Incorrect reason for notification.
C. CORRECT	In accordance with the Normal Operating Procedure, RP is to be notified whenever the ventilation alignment in the Aux Building has been altered to ensure proper monitoring of radiation levels.
D. INCORRECT	Wrong person notified. Incorrect reason for notification.

OBJECTIVE: RO-C-0280040501-E3
REFERENCE: 12-OHP-4021-028-011, Attachment 1, Step 2.1.

KA - 194001 2.3.13
Generic
Radiation Control
Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.
RO - 3.4 SRO - 3.8
CFR - 41.12 / 43.4 / 45.9 / 45.10

KA Justification - Question requires operator knowledge of action required to be taken for radiological safety when configuring ventilation systems.

Original Question # -AUDIT RO23 RO-36, RO25 Audit-73, CM-1132, NRCAUDIT07-0452
Original Question KA - Generic 2.3.10

72. 072 002/RO/OK/DIRECT/NRC EXAM 2006-064-13/194001 2.3.15/RO-C-02200-E8/NRCAUDIT07-0837/F/3

Which ONE of the following will AUTOMATICALLY stop the selected Monitor Tank pump during a liquid release to the Unit 2 Circulating Water System?

A. HIGH flow alarm on Liquid Waste Sample Flow channel RFS-1010.

B. ALERT alarm on Liquid Waste Effluent channel RRS-1001.

C. Loss of all Unit 2 Circulating Water pumps.

D. HIGH alarm on Liquid Waste Local Area channel RRA-1003.

ANSWER: A

A. CORRECT	HIGH or LOW sample flow on RFS-1010 will cause the monitor tank pumps to trip. Channel Failure on RRS-1001 or RRR-1002 or a high radiation alarm on RRS-1001 will also cause the pumps to trip.
B. INCORRECT	An "Alert" alarm will NOT cause the pumps to trip.
C. INCORRECT	A loss of all CW will NOT automatically cause the pumps to trip, but 2-RRV-286, Liquid Waste Effluent to Unit 2 CW Discharge valve will auto close.
D. INCORRECT	High local area alarm will NOT cause the monitor tank pumps to trip.

OBJECTIVE: RO-C-02200-E8
REFERENCE: 12-OHP-4024-139, Annunciator #139 Response: Eberline Radiation, # 18, Radioactive Liquid Effluent Monitor RRS-1000.

KA - 194001 2.3.15
Generic
Radiation Control
Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.
RO - 2.9 SRO - 3.1
CFR - 41.12 / 43.4 / 45.9

KA Justification - Question requires operator knowledge of the actions resulting from a radiation monitor alarm.

Original Question # - NRC EXAM 2006-064-13, RO23 AUDIT-018-4, COOK02-075, NRCAUDIT07-0837, Original Question KA - G 2.3.11, 068000 - K6.10

73. 073 024/RO/OK/DIRECT/CM-3256/194001 2.4.6/RO-C-EOP09-E11/CM-3256/F/3

Which ONE of the following is correct concerning RCP trip criteria and/or RCP operation during a Small Break LOCA event?

A. The purpose of the trip criteria is to minimize RWST depletion that would result from the increased mass loss caused by RCP operation.

B. The RCPs are tripped because natural circulation flow is a more effective core heat removal mechanism when considering two-phase flow.

C. If the RCPs trip prior to break uncovery, more core uncovery will occur than if they are tripped later in the event.

D. If the RCPs continued to operate for the duration of the event, sufficient two-phase flow would be available to adequately remove heat from the core.

ANSWER: D

A. INCORRECT	While operation of the RCPs would lead to more mass loss, the purpose is not to conserve RWST inventory. The RWST inventory will be available in the sump.
B. INCORRECT	Two phase heat removal is more effective but can not be guaranteed.
C. INCORRECT	RCP trip prior to break uncovery shows very little impact on total mass loss or core uncovery.
D. CORRECT	If the RCPS remain running heat removal is more effective. Two phase heat removal is more effective but can not be guaranteed.

OBJECTIVE: RO-C-EOP09-E11
REFERENCE: RO-C-EOP09 pg. 129

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 - Question requires operator knowledge of the impact that the RCP trip criteria has on event mitigation.

Original Question # - CM-3256
Original Question KA - 1940012406

74. 074 002/RO/OK/NEW/NEW/194001 2.4.31/RO-C-ADM02-E25/RO-C-ADM02-E25-1/F/2

Which ONE of the following describes the significance of the "GREEN" annunciators in the control room based on the OHI-4000 (Conduct of Operations, Alarm Response) Annunciator Priority System?

The Green Lens alarms ...

A. mean that compensatory actions may be required.

B. signify those alarms which are generated by local panels.

C. signify those alarms which are expected on a shiftly basis and do not need to be announced.

D. indicate alarms which seal-in and require the Reset button to clear once the condition has cleared.

ANSWER: B

A. INCORRECT	A Red "C" on the lens denotes compensatory actions required.
B. CORRECT	Green lens indicates local panel alarms.
C. INCORRECT	Expected alarms do need to be announced but may not require ARP usage. Plausible, as candidate may confuse green with expected.
D. INCORRECT	Seal-in alarms have a slash in the lower corner.

OBJECTIVE: RO-C-ADM02-E25
REFERENCE: OHI-4000 Attachment 1, Alarm Response, Section 3.9

KA - 194001 2.4.31
Generic
Emergency Procedures/Plan
Knowledge of annunciator alarms, indications, or response procedures.
RO - 4.2 SRO - 4.1
CFR - 41.10 / 45.3

KA Justification - Question requires operator knowledge of the coding of the annunciators and their meaning.

Original Question # - NEW NRC Exam 2012-074
Original Question KA - N/A

75. 075 007/RO/OK/DIRECT/NRC EXAM 2002-006/194001 2.4.49/RO-C-EOP04-E13/CM-39479/H/3

The Unit 2 reactor failed to automatically trip when the Reactor Coolant Pumps tripped.

The following conditions exist after a MANUAL Turbine Trip is attempted per OHP-4023-FR-S-1, Response to Nuclear Power Generation/ATWS step 3:

- Turbine Stop Valve Closed Status Lights - 1, 2, and 3 Lit
- Turbine Stop Valve Closed Status Lights - 4 NOT Lit
- MAIN TURBINE STOP VALVE CLOSED alarm - Lit
- AMSAC INITIATED alarm - Lit

Which ONE of the following is the NEXT action the operator is required to take?

A. Shut the Main Steam Stop Valves.

B. Actuate ATWS Turbine Runback.

C. Verify AFW Pumps running.

D. Manually actuate AMSAC.

Answer: B

A. INCORRECT	Closing the Main Steam Stop valves is only performed if a manual load reduction does not work.
B. CORRECT	The turbine is verified tripped by checking all 4 status lights closed. The alarms are lit based on 1 stop valve closed and the AMSAC initiated. Since the turbine is not tripped, 2-OHP-4023-FR-S-1 requires that load be manually reduced.
C. INCORRECT	Checking the AFW Pumps running is step 4 but not the next action since the turbine trip has not been verified.
D. INCORRECT	Step 2 of 2-OHP-4023-FR-S-1 actuates AMSAC. This is NOT performed in the turbine trip verification step as it is in the 2-OHP-4023-E-0, Reactor Trip or Safety Injection, procedure.

OBJECTIVE: RO-C-EOP04-E13
REFERENCE: 2-OHP-4023-FR-S-1, Response to Nuclear Power Generation/ATWS Step 3

KA - 194001 2.4.49
Generic
Emergency Procedures/Plan
Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
RO - 4.6 SRO - 4.4
CFR - 41.10 / 43.2 / 45.6

KA Justification - Question requires operator evaluate a turbine trip per FR-S.1 and determine correct actions to take to complete the trip.

Original Question # - CM-39479, NRC Exam 2002-006, RO23 Audit-5
Original Question KA - 012000 K3.02, 000029 - EA1.13

76. 076 004/SRO/OK/NEW/NEW/002000 A2.04/RO-C-EOP11-E7/RO-C-EOP11-E7-1/H/3

Given the following conditions:

- Unit 2 was operating at 100% power.
- The West Centrifugal Charging Pump (CCP) was tagged out due to shaft misalignment.
- Following a feedwater line break outside containment the crew encountered complications due to a loss of Auxiliary Feedwater (AFW).
- The operating crew responded in accordance with the Emergency Operating Procedures and has entered OHP-4023-FR-H.1, Response to Loss of Secondary Heat Sink.

- When checking if Bleed and Feed is required, the following is noted:
  - The East CCP just Tripped.
  - Reactor Coolant System (RCS) pressure is currently 2280 psig.
  - All Three Pressurizer PORVs are available.
  - Steam Generator (SG) levels are:

SG #1:	38% wide range
SG #2:	5% wide range
SG #3:	31% wide range
SG #4:	38% wide range

Which ONE of the following identifies the requirement for RCS Bleed and Feed and the subsequent required actions?

A. RCS Bleed and Feed cooling must NOT be initiated. Continue with OHP-4023-FR-H.1 steps to perform an immediate secondary depressurization to inject condensate pump flow.

B. RCS Bleed and Feed cooling must NOT be initiated. Transition to OHP-4023-FR-C.1, Response to Inadequate Core Cooling immediately.

C. Immediately initiate RCS Bleed and Feed and then transition to OHP-4023-FR-C.1, Response to Inadequate Core Cooling.

D. Immediately initiate RCS Bleed and Feed and then continue with OHP-4023-FR-H.1 steps to restore a secondary heat sink.

ANSWER: D

A. INCORRECT	Bleed and feed is required immediately if both high head CCPs are lost due to the concern over the amount of injection flow possible. Plausible since SG levels are acceptable.
B. INCORRECT	Bleed and feed is required immediately if both high head CCPs are lost due to the concern over the amount of injection flow possible. Plausible since operator may confuse concern over loss of CCP with requirement for FR-C.1.
C. INCORRECT	Bleed and feed is required immediately since both high head CCPs are lost. Plausible since loss of CCPs may also lead to requirement for FR-C.1.
D. CORRECT	Bleed and feed is required immediately since both high head CCPs are lost.

OBJECTIVE: RO-C-EOP11-E7, RO-C-EOP11-13
REFERENCE: 12-OHP-4023-FR-H.1 PSBD, Step 2; 2-OHP-4023-FR-H.1, Step 2.

KA - 002000 A2.04
Reactor Coolant System (RCS)
Ability to (a) predict the impacts of the following malfunctions or operations on the RCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Loss of heat sinks
RO - 4.3 SRO - 4.6
CFR - 41.5 / 43.5 / 45.3 / 45.5

KA Justification - Question requires candidate to evaluate the impact to the plant due to the loss of heat sink and high head injection and select the correct mitigation strategy and procedure.

Original Question # - New NRC EXAM 2012-076
Original Question KA - N/A

77. 077 002/SRO/OK - ATTACHMENT/DIRECT/NRC EXAM 2004-121/004000 2.2.22/RO-C-03200-E20/RO26-0172/H/4

Given the following conditions on Unit 2:

- Unit is at 100% with Emergency Diesel Generator 2AB out of service due to contaminated fuel oil.
- The diesel was declared inoperable at 1100 on 8/15/12.
- The Supplemental Diesel Generators are NOT available.
- At 1600 on 8/15/12, the plant experiences a trip due to a spurious reactor trip signal generated during Instrument Maintenance testing.
- At 2200 on 8/15/12, while maintaining the plant in Hot Standby, Annunciator Panel 204 Drop 86 CCW FROM EAST CCP PUMP FLOW LOW, alarms.
- Investigation shows CCW flow to East CCP has been lost due to an apparent valve stem/disc separation. CCW flow to the West CCP is normal.

Which ONE of the following describes the status for plant startup?

Attachment Provided.

Modes 1 and 2 may ...

A. be entered. Power operations may continue as long as Emergency Diesel Generator 2AB is restored to service by 1100 on 8/18/12.

B. be entered. Power operations may continue as long as East CCP is restored to service by 2200 on 8/18/12.

C. NOT be entered and must be in Cold Shutdown NO later than 0900 on 8/17/12.

D. NOT be entered and must be in Cold Shutdown NO later than 1500 on 8/17/12.

ANSWER: D

A - INCORRECT. Both CCPs are Inoperable. Plausible if impact to opposite train/associated CCP is missed. This is 72 hours from EDG inoperable.

B - INCORRECT. Both CCPs are Inoperable. Plausible if impact to opposite train/associated CCP is missed. This is 72 hours from East CCP Inoperable.

C - INCORRECT. This time limit excludes 6 hours since the plant is already in Mode 3. TS 3.0.3 just requires 37 hours to Mode 5.

D - CORRECT. With 2AB EDG inoperable, for the West CCP to be considered operable its normal power source must be operable AND the East CCP must be operable. The given condition results in East CCP being inoperable. This places the plant in the requirements of TS 3.8.1 Action B.3 requiring restoration in 4 hours or West CCP is also Inoperable. TS 3.5.2 Action C (2 ECCS Pumps <100% single train flow) requires TS 3.0.3 to be implemented immediately which requires Cold Shutdown in 37 hours. (6 hours to Hot Standby is lost).

OBJECTIVE: RO-C-03200-E20
REFERENCE: TS 3.8.1 Action B, TS 3.5.2, 3.0.2, 3.0.3, 3.0.6

Attachment Provided: UNIT 2 TS Section 3.0, 3.5.2, and 3.8.1

KA - 004000 2.2.22
Chemical and Volume Control System (CVCS)
Equipment Control
Knowledge of limiting conditions for operations and safety limits.
RO - 4.0 SRO - 4.7
CFR - 41.5 / 43.2 / 45.2

KA Justification - K/A Match and **SRO Only**:
Applicant is presented with a equipment configuration to determine the OPERABILITY status of plant equipment in accordance with TS LCOs (SRO level knowledge, since it is application of Tech. Spec. rules for mode changes).

Original Question # - RO27 Audit -96, Cook NRC Exam 2004-121-1 (SRO#94), RO26 Audit-83 , INPO 21616 KEWAUNEE 2002, NRCAUDIT07-0066, RO26-0172
Original Question KA - 000026 2.2.42, 194001 2.2.42

78. 078 002/SRO/OK/NEW/NEW/008000 A2.09/RO-C-00300-E19/RO-C-00300-E19-1/H/3

Given the following plant conditions:

- Unit 2 is operating at 100% power.
- Ann. 209, Drop 8, LETDOWN HX OUTLET TEMP HIGH is LIT.
- Letdown HX Outlet temperature is 135°F and rising slowly (and has been verified locally).
- CRV-470, Letdown Hx Temp Ctrl shows 100% demand and will **NOT** transfer to MANUAL.

Which ONE of the following describes this failure and the required actions to mitigate this event?

CRV-470, Letdown Heat Exchanger Temperature Control Valve has failed...

A. CLOSED. If temperature continues to rise, direct the operator to place QRV-302, Cold Letdown Path Select to the RC Filter Position, Close the Reactor Coolant Normal Letdown Isolation Valves, and Place Excess Letdown in service per 2-OHP-4021-003-001, Letdown, Charging And Seal Water Operation.

B. CLOSED. Raise Charging flow by adjusting QRV-200, Charg Hdr Press Ctrl, and QRV-251, CCP Disch Flow Ctrl. and enter 2-OHP-4022-016-001, Malfunction Of The CCW System to address this failure.

C. OPEN. Place divert valve QRV-303, VCT/ Holdup Tk Inlet Selector, to CVCS Hold Up Tanks and borate the RCS per 2-OHP-4021-005-002, Operation of the Boric Acid Blender to counteract the dilution effects.

D. OPEN. Place divert valve QRV-303, VCT/ Holdup Tk Inlet Selector, to CVCS Hold Up Tanks and borate the RCS per 2-OHP-4021-005-007, Operation Of Emergency Boration Flow Paths to counteract the dilution effects.

ANSWER: A

A. CORRECT. Rising temperature indicates that the cooling valve has failed CLOSED. The demins will need to be bypassed, letdown removed from service, and excess letdown used.

B. INCORRECT. Raising Charging flow will provide some additional cooling but also raise PZR level. The CCW procedure will only address a loss of all CCW or CCW system malfunction (plausible since it is listed in the ARP).

C. INCORRECT. Valve failure direction is incorrect, plausible if HX used bypass flow to control temp. If the Valve had failed open, this would be an expected response (colder water causes demin to absorb boron).

D. INCORRECT. Valve failure direction is incorrect, plausible if HX used bypass flow to control temp. If the Valve had failed open, this would be an expected response (colder water causes demin to absorb boron).

OBJECTIVE: RO-C-00300-E19; RO-C-00300-E7; RO-C-01600-E4
REFERENCE: 2-OHP-4024-209, Drop 8; SOD-00300-001

KA - 008000 A2.09
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:
Results of excessive exit temperature from the letdown cooler, including the temperature effects on ion-exchange resins
RO - 2.3 SRO - 2.8
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Question requires knowledge of the cause of the failure and impacts as well as detailed procedural mitigation strategies.

Original Question # - NRC EXAM 2012-078
Original Question KA - N/A

79. 079 002/SRO/OK/DIRECT-REPEAT/NRC EXAM 2008-87/010000 A2.02/RO-C-AOP0330412-E2/2008NRC-0620/H/3

Given the following plant conditions on Unit 2:

- Reactor power is 12%.
- The controlling Pressurizer (PRZ) Pressure Channel slowly fails high.
- The RO takes manual control of PRZ Pressure Master Controller and Lowers demand.
- All PRZ heaters have energized.
- RCS pressure is 2075 psig and slowly lowering.
- You notice that NRV-163 (PRZ spray) is failed OPEN.
- When the RO places NRV-163 in manual it will NOT close.

Which ONE of the following is the proper sequence of actions to stop the pressure reduction?

A. Trip RCP #3.
Dispatch an AEO to locally isolate Spray Valve NRV-163.

B. Reduce Power to 8%.
Trip RCPs #3 and #4.
Dispatch an AEO to locally isolate Spray Valve NRV-163.

C. Trip RCP #3.
Go to OHP-4023-E-0, Reactor Trip Or Safety Injection.

D. Manually trip the reactor.
Go to OHP-4023-E-0, Reactor Trip Or Safety Injection.
Trip RCPs #3 and #4.

ANSWER: D

A. INCORRECT. Three loop operation is not allowed per the license, but plausible since the reactor would not trip at this power on the loss of a single RCP.

B. INCORRECT. Three loop operation is not allowed per the license. Plausible since 10% power also blocks 2 loop loss of flow trip.

C. INCORRECT. Per operating practices, the Reactor is tripped first and then the RCP. Plausible since a reactor trip and pump trip is required.

D. CORRECT. Three loop operation is not allowed. The RCP#3 & 4 must be stopped to stop the spray flow. Therefore the Reactor must be manually tripped and then the RCPs tripped.

OBJECTIVE: RO-C-AOP0330412-E3, RO-C-AOP0330412-E2
REFERENCE: 2-OHP-4024-207, Annunciator #207 Drop 61; 2-OHP-4023-ES-0-1, Reactor Trip Response step 5

KA - 010000 A2.02
Pressurizer Pressure Control System (PZR PCS)
Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Spray valve failures
RO - 3.9 SRO - 3.9
CFR - 41.5 / 43.5 / 45.3 / 45.13

KA Justification - Question requires candidate to identify actions required and their order based on PRZ Spray Valve Failure.

Original Question # - NRC EXAM 2008-87, RO24 AUDIT-079-11, Cook NRC Exam 2002-080 (#66/65), INPO - MODIFIED 19458
Original Question KA - 010000 A2.02

80. 080 002/SRO/OK/DIRECT/NRC EXAM 2006-089-6/012000 2.2.37/RO-C-01100-E11/CM-7876/H/3

Unit 1 is in MODE 3 preparing for a reactor startup. All shutdown bank rods are withdrawn.

I&C informs you that MPC-254 (Main Turbine 1st stage pressure transmitter) must be placed in the test (failed high) position in order to perform work on the transmitter.

Which ONE of the following correctly describes your response to this request and the reason?

A. Allow Testing. MPC-253 is still indicating properly, so P-13 and P-7 will indicate correctly.

B. Do NOT Allow Testing. The reactor trips associated with permissive P-7 would be unblocked.

C. Do NOT Allow Testing. The Steam Dumps will close when MPC-254 is placed in the Failed Condition.

D. Allow Testing, after placing AMSAC to the Bypass Condition.

ANSWER: B

A. INCORRECT	2/2 channels below 10% are required for P-13 & P-7.
B. CORRECT	P-13 is actuated by 2 of 2 Impulse channels below 10. Failing 1 channel high will remove P-13 which will remove the P-7 Blocks.
C. INCORRECT	Steam dumps are in steam Pressure Mode during these conditions. The dumps would close if in Tave Mode.
D. INCORRECT	AMSAC requires 2/2 Channels >40% to enable.

OBJECTIVE: RO-C-01100-E11, RO-C-01100-E13
REFERENCE: 1-OHP 4022.013.016, Attachment B

KA - 012000 2.2.37
Reactor Protection System
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 - Question requires SRO to determine the availability and impact to RPS of placing a channel in test.

Original Question # - CM-7876, NRCAUDIT07-0767, NRC EXAM 2006-089-6, Master Bank 01011C0004-3

Original Question KA - 016 A2.01, 016000 2.2.18

81. 081 004/SRO/OK - ATTACHMENT/NEW/NEW/017000 A2.01/RO-C-01301-E12/RO-C-01301-E12-1/H/3

Unit 2 is operating at 100% power. The 43-TSAT-2 Thermocouple Selector Switch is selected to use the Auctioneering function.

A "Short" has caused ALL of the Train A and Train B Quadrant 1 Core Exit Thermocouples (CET) to Fail.

Which ONE of the following describes the operability and the required actions to address the associated Technical Specifications?

Attachment Provided

The CET Quadrant 1 Instrumentation is INOPERABLE and the RCS Subcooling Margin Monitor (SMM) is ...

A. OPERABLE. Restore one train of Quadrant 1 CETs within 7 days or be in Mode 3 within 6 hours and Mode 4 within 12 hours.

B. OPERABLE. Operation is allowed for up to 30 days or submit a Post Accident Monitoring Report detailing the restoration plan as directed by Specification 5.6.6 .

C. INOPERABLE. Select a Quadrant 2,3, or 4 CET to feed the RCS SMM Immediately AND Restore one train of Quadrant 1 CETs within 7 days or be in Mode 3 within 6 hours and Mode 4 within 12 hours.

D. INOPERABLE. Restore one train of Quadrant 1 CET (which also restores the RCS SMM) within 30 days or submit a Post Accident Monitoring Report detailing the restoration plan as directed by Specification 5.6.6.

ANSWER: A

A. CORRECT	A CET with a "Short" will indicate LOW (200^{o}F) . There are eight thermocouple signals to the SMM. The operator can select high-select or one of seven of the eight thermocouple signals in order to perform the margin calculation. Since the Stem states that the Switch is in the Auctioneering Function, the failure does not impact the Saturation meter indication. TS. 3.3.3 Action D (two trains) is applicable. Failure to Complete D requires Condition E & F.
B. INCORRECT	Both Trains do need to be restored within 30 days but if one is not back within 7 days a shutdown is required.
C. INCORRECT	SMM is still operable, another CET does not need to be selected.
D. INCORRECT	SMM is still operable and both Trains do need to be restored within 30 days but if one is not back within 7 days a shutdown is required.

OBJECTIVE: RO-C-01301-E12; RO-C-01301-E11
REFERENCE: TS 3.3.3 PAM Instrumentation

Attachment Provided : Unit 2 TS 3.3.3 PAM Instrumentation

KA - 017000 A2.01
In-Core Temperature Monitor (ITM) System
Ability to (a) predict the impacts of the following malfunctions or operations on the ITM System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
Thermocouple open and short circuits
RO - 3.1 SRO - 3.5
CFR - 41.5 / 43.5 / 45.3 / 45.5

KA Justification - Question requires operator to determine the impact of the CET failures (fail low and effect on SMM) and then use TS to determine the correct limits and actions for restoration.

Original Question # NRC EXAM 2012-081
Original Question KA - N/A

82. 082 007/SRO/OK/DIRECT/RO25 AUDIT-87/041000 2.4.45/RO-C-NOP6-E2/NRCAUDIT07-1000/H/3

Given the following plant conditions:
Unit 1 is performing OHP-4021-002 Reactor Start-Up .

- Reactor Power is at 3% power .
- UPC-101, Bypass Steam Header Pressure, has just failed HIGH.
- Annunciator Panel 111 Drop 2 LOOPS 1 2 3 4 TAVG LOW-LOW has just alarmed
- The Operator placed the Steam Dump Pressure Control in Manual and reduced demand to 0%.
- All Steam Dumps currently indicate closed EXCEPT URV-124 which indicates full open.
- RCS temperature is 538˚F and lowering.

Which ONE of the following describes the required actions?

A. Place the "A" and "B" "STEAM DUMP INTLK BYP" switches to "OFF/RESET" to close the Steam Dump Valve and restore RCS Temperature to > 541˚F within 30 minutes.

B. Dispatch operator to isolate the steam dump valve locally and restore RCS Temperature to > 541˚F within 30 minutes.

C. Place the "A" and "B" "STEAM DUMP INTLK BYP" switches to "OFF/RESET" to close the Steam Dump Valve and slowly withdraw Control Bank D until Tave is > 541˚F.

D. Immediately trip the reactor and go to OHP-4023-E-0, Reactor Trip or Safety Injection.

ANSWER: D

A. INCORRECT Steam Dumps receive a close signal when Tave is < 541˚F. so they should already be closed. OHP-4021-002 Reactor Start-Up Step 3.12 states that criticality shall NOT be maintained with Tave < 539˚F. So the Reactor Should be Tripped.

B. INCORRECT OHP-4021-002 Reactor Start-Up Step 3.12 states that criticality shall NOT be maintained with Tave < 539˚F. So the Reactor Should be Tripped.

C. INCORRECT Steam Dumps receive a close signal when Tave is < 541˚F. so they should already be closed. OHP-4021-002 Reactor Start-Up Step 3.12 states that criticality shall NOT be maintained with Tave < 539˚F. So the Reactor Should be Tripped.

D. CORRECT OHP-4021-002 Reactor Start-Up Step 3.12 states that criticality shall NOT be maintained with Tave < 539˚F. So the Reactor Should be Tripped. Atmospheric dumps have local isolation valves. The Steam Dumps receive a close signal when Tave is < 541˚F. This can be bypassed for Group 1 only. TS 3.4.2 requires Tave to be > 541˚F when the reactor is critical. This must be restored within 30 minutes.

OBJECTIVE: RO-C-NOP6-E2, RO-C-NOP6-T2
REFERENCE:1-OHP-4021-001-002 Reactor Start-Up step 3.12

KA - 041000 2.4.45
Steam Dump System (SDS) and Turbine Bypass Control
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 - Question requires the operator to determine the significance of the LOW - LOW alarm (dumps should be closed and Plant is below TS limit) and direct the correct actions.

Original Question # - NRCAUDIT07-1000, RO25 AUDIT-87
Original Question KA - 041000 A2.02

83. 083 005/SRO/OK - ATTACHMENT/DIRECT/CM-1002/076000 A2.01/RO-C-AOP0480412-E3/CM-1002/H/3

Given the following conditions:

- Unit 1 is at 25% power with the Steam Dump System in the Tavg Mode.
- The South NESW pump out of service for maintenance.
- Annunciators associated with low NESW header pressure are received.
- The crew has closed 1-WMO-906 (Pump Discharge Hdr Xtie to Unit 2).
- NESW Header pressure is observed to be less than normal.
- The Unit 1 Turbine Tour AEO reports water spilling from the common 10" supply line to the Unit 1 Main Turbine Lube Oil Coolers. (Leakage is being collected in floor drains).

Which ONE of the following describes the correct course of action?

Prompt action should be taken to ...

Attachment Provided

A. Trip the reactor and go to 1-OHP-4023-E-0, Reactor Trip and Safety Injection.
Transition to 1-OHP-4023-ES-0.1, Reactor Trip.
Trip the Main Turbine and Main Feed Pumps and then break condenser vacuum.

B. Trip the Main Turbine and Main Feed Pumps
Break condenser vacuum.
Stop Main Turbine Lube Oil Pumps

C. Stop and lockout the Unit 1 North NESW pump, and close its suction and discharge valves.
Trip the reactor and go to 1-OHP-4023-E-0, Reactor Trip and Safety Injection.
Transition to 1-OHP-4023-ES-0.1, Reactor Trip.
Stop all but one RCP

D. Request Unit 2 implement 2-OHP-4022-020-001, NESW System Loss/Rupture.
Re-open 1-WMO-906 (Pump Discharge Hdr Xtie to Unit 2).
Return to 1-OHP-4021-001-006, Power Escalation

ANSWER: B

A. INCORRECT A RX trip is not required since RX power is 25% (<P8). Step 38 RNO
B. CORRECT Since power is 25% (<P8). Step 39-45 directs these actions
C. INCORRECT These are actions directed by Step 4 for Flooding.
D. INCORRECT These actions would be indicative of a leak on Unit 2 (Step 3 RNO)

OBJECTIVE: RO-C-AOP0480412-E2, RO-C-AOP0480412-E3
REFERENCE: 1-OHP-4022-020-001, NESW System Loss/Rupture, Step 38-45.

Attachment Provided: **1-OHP-4022-020-001, NESW System Loss/Rupture**

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

KA Justification - Question requires knowledge of impact to the plant due to a loss of NESW to major components and the required procedural direction.

Original Question # - CM-1002
Original Question KA - SF4.076.A2.02

84. 084 002/SRO/OK/DIRECT/NRC EXAM 2002-027/000007 2.2.38/RO-C-ES0-1-E11/RO26-0171/H/3

Given the following plant conditions:

- A Reactor trip from 100% power has occurred on Unit 1.
- OHP-4023-E-0, Reactor Trip Or Safety Injection, is being implemented.
- Containment pressure is 0.6 psig and stable.
- SG NR Levels are offscale low.
- RCS pressure is 2150 psig and lowering.
- Control Rod H-8 is indicating 32 steps.
- All systems responded normally to actuation signals.

Which ONE of the following actions must be taken?

A. Transition to OHP-4023-ES-0-1, Reactor Trip Response, and initiate boration for the stuck rod.

B. Transition to OHP-4023-ES-0-1, Reactor Trip Response. Rod H-8 condition is expected so boration is not required for a stuck rod.

C. Initiate Safety Injection and continue with OHP-4023-E-0, Reactor Trip Or Safety Injection, as pressurizer pressure is too low.

D. Initiate Safety Injection and continue with OHP-4023-E-0, Reactor Trip Or Safety Injection, as Steam Generator levels are too low.

ANSWER: A

A. CORRECT. Rod H-8 and all rods should be less than 10 steps. Plant conditions do not require a Safety Injection. Boration for the stuck rod is initiated in OHP-4023-ES-0-1, Reactor Trip Response .

B. INCORRECT. Rod H-8 is expected to be less than 10 steps on Unit 1. Unit 2 Rod H-8 previously was required to be less than 35 steps.

C. INCORRECT. RCS pressure will decrease post trip and it is still above the SI setpoint.

D. INCORRECT. SG levels will decrease to offscale low post trip. AFW will recover levels.

OBJECTIVE: RO-C-EOP03-E23, RO-C-ES0-1-E11
REFERENCE: 1-OHP-4023-E-0, Reactor Trip Or Safety Injection, Step 4;
1-OHP-4023-ES-0-1, Reactor Trip Response, Step 6.

KA - 000007 2.2.38
Reactor Trip - Stabilization
Equipment Control
Knowledge of conditions and limitations in the facility license.
RO - 3.6 SRO - 4.5
CFR - 41.7 / 43.1 / 45.13

K/A Justification - Requires the ability to interpret the reactor and based on the indications, implement EOP actions as required for reactor trip status and the maintenance of SDM as required by the license.

Original Question # - RO26-0171, RO26 AUDIT 77, Cook NRC Exam 2002-027-42 (SRO#25)Original Question KA - 000007 EA2.06

NOTE: Previously on Unit 2, correct answer would be "Transition to 02-OHP-4023-ES-0-1. Rod H-8 condition is expected so boration is not required for a stuck rod." - This long term condition has been corrected and is no longer a difference.

85. 085 002/SRO/OK/MODIFIED/NRC EXAM 2010-21/000022 AA2.01/RO-C-ADM02-E36/RO-C-ADM02-E36-1/H/4

Given the following conditions on Unit 2:

- Reactor is at 100% power.
- All control systems are in normal alignment.
- Letdown flow is 75 gpm on QFI-301, Letdown Flow Indicator.

The following parameters are now noted on the CVCS system:

- Seal Return Flows are 3 gpm per Reactor Coolant Pump.
- Charging flow is 94 gpm and rising.
- QTA-160, Regen HX Outlet Temp - Letdown, has risen 25°F from its steady state value.
- Volume Control Tank level is 33% and lowering.
- Pressurizer level is 55% and lowering slowly.
- Reactor Coolant System (RCS) average temperature is 574°F and stable.

Which ONE of the following describes the correct leak location AND associated leakage monitoring requirements?

The Leakage is from the ...

A. letdown line between the letdown isolation valves and the orifices valves. This leakage must be monitored as post accident recirculation flowpath leakage, as required by TS 5.5.2, Leakage Monitoring Program.

B. charging line between the flow indicator and Containment. This leakage is required to be monitored as RCS leakage, as required by TS 3.4.13 RCS Operational LEAKAGE.

C. letdown line between the letdown isolation valves and the orifices valves. This leakage is required to be monitored as RCS leakage, as required by TS 3.4.13 RCS Operational LEAKAGE.

D. charging line between the flow indicator and Containment. This leakage must be monitored as post accident recirculation flowpath leakage, as required by TS 5.5.2, Leakage Monitoring Program.

ANSWER: D

A - INCORRECT. Wrong location for leak. This leakage would cause a rise in letdown flow through the Regen HX and a lowering on QTA-160, Regen HX Outlet Temp - Letdown. However, this leakage would be inside Containment and would not apply to the TS 5.5.2.

B - INCORRECT. Correct location for leak but leakage outside containment would be TS 5.5.2.

C - INCORRECT. Wrong location for leak. Leak is in the charging header. This leakage is required to be monitored as per TS 5.5.2.

D - CORRECT. Correct location for leak. This leakage would cause a temperature rise on QTA-160, Regen HX Outlet Temp - Letdown due to less charging flow to cool the letdown exiting the RHX. Since charging flow increased from a normal value of 87 GPM to 94 GPM with no other changes the leak rate is approximately 7 GPM and will required isolation of the charging header to isolate the leak.

OBJECTIVE: RO-C-ADM02-E36; RO-C-AOP0160412-E1; RO-C-AOP0160412-E3
REFERENCE: SOD-00300-001; OHI-4032 (Note US/SM responsibilities)

KA - 000022 AA2.01
Loss of Reactor Coolant Makeup
Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup:
Whether charging line leak exists
RO - 3.2 SRO - 3.8
CFR - 43.5 / 45.13

KA Justification - Question requires operator to determine where the leak exists and the required monitoring per Section 5 of Tech Specs.

Original Question # - Modified from NRC 2010 Exam-21, RO26 AUDIT-4, SEQ2007, RO26-0005
Original Question KA - 004000 2.4.47, 004000 K6.07

Modified by Changing stem Letdown Flow 120 to 75, Charging 137 to 94, and letdown temp RISING by 25 °F vs. lowering. This changes leak location to make D location correct. Changed last sentence in stem from "may" to "is." Changed distractor B leak location (was on RHX). Changed distractor C (to LD leak and Operational Leakage) to balance distractors. Modified all explanations.

86. 086 002/SRO/OK/MODIFIED/NRC EXAM 2010-80/000032 AA2.06/RO-C-01300-E21/RO-C-01300-E21-1/H/3

Given the following conditions on Unit 2:

- A reactor startup is in progress with the reactor just critical.
- Intermediate Range Power indication has just come online and is 3 x 10^{-11} amps.
- Source Range (SR) nuclear instrumentation channel (N-31) Instrument Power Fuse Blows.

Which ONE of the following actions, if any, is required?

A. NO action required, source ranges are NOT required to be operable.

B. VERIFY Reactor trip and enter OHP-4023-E-0, Reactor Trip or Safety Injection. Suspend all operations involving positive reactivity changes and restore both SR channels to operability within 48 Hours.

C. Conduct a reactor shutdown and restore both SR channels to operability prior to the next startup.

D. VERIFY Reactor trip and enter OHP-4023-E-0, Reactor Trip or Safety Injection. Place N31 in the trip position within 1 hour and restore both SR channels to operability within 48 Hours.

ANSWER: B

A - INCORRECT. The Reactor will trip. Plausible, as TS 3.3.1 (Instrumentation) establishes that above P-6, the SR NIs are not required by TS and will shortly be de-energized by procedure, since there are no TS implications, the startup may proceed.

B - CORRECT. The Reactor will trip. The SR channel requires restoration within 48 hours and suspension of positive reactivity.

C - INCORRECT. The reactor will trip. If power was >P-6 a plant shutdown would not be required. 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. The Reactor will trip with one SR Channel below P-6. However, the SR channel does not need to be placed in the tripped condition.

LESSON PLAN/OBJ: RO-C-01300-E21
REFERENCE: TS 3.3.1, RTS Instrumentation, Table 3.3.1-1

KA - 000032 AA2.06
Loss of Source Range Nuclear Instrumentation
Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation:
Confirmation of reactor trip
RO - 3.9 SRO - 4.1
CFR - 43.5 / 45.13

KA Justification - Requires an evaluation of the plant status due to a loss of source range instruments below the P-6 setpoint and the TS actions required.

Original Question # - Modified from NRC 2010 EXAM-80, RO24 AUDIT-021-7
Original Question KA - 000032 2.1.7, APE.032GEN2.1.20

Modified Stem by lowering IR power to <P6, changed SR failure to Blown fuse (vs. Fail low). This changed answer from no action required (A) to Trip (B). Added TS requirements to B and changed D to be trip and incorrect TS requirements.

Removed Supplied Reference.

87. 087 003/SRO/OK - ATTACHMENT/DIRECT/RO26 AUDIT-62/000054 AA2.06/RO-C-EOP11-E18/RO26-0077/H/3

The crew has just completed actions of 1-OHP-4023-FR-H.1, Response to Loss of Secondary Heat Sink, that established bleed and feed.

Given the following plant conditions:

- SG #3 is faulted with ZERO indicated level on SG Wide Range Level.
- SG #1, 2, and 4 are intact and SG Wide Range levels are 14%.
- Core Exit Thermocouples are 570°F (hottest) and slowly lowering.
- RCS Hot Leg temperatures are 565°F and slowly lowering.
- RCS Wide Range Pressure is 1200 psig and stable
- Containment pressure is 0.8 psig and rising.
- The Turbine-driven AFW Pump was just made available.

What actions are taken to establish a secondary heat sink and ensure core heat removal?

Attachment Provided

A. Initiate feed flow to SG #1, 2, **OR** 4 at less than 50 X10^3 lbm/hr.
Raise feed flow as required after SG Wide Range level is > 17%.
Terminate bleed and feed after SG Narrow Range level is above 14%.

B. Initiate feed flow to SG #1, 2, **OR** 4 at the maximum rate until RCS Subcooling is > 40°F.
Terminate bleed and feed after SG Narrow Range level is above 14%.

C. Initiate feed flow to SG #3 at the maximum rate until RCS Subcooling is > 40°F.
Initiate feed flow to SG #1, 2, **OR** 4 at less than 50 X10^3 lbm/hr after RCS Subcooling is > 40°F.
Terminate bleed and feed after any SG Wide Range level is > 17%.

D. Initiate feed flow to SG #1, 2, **AND** 4 at less than 50 X10^3 lbm/hr.
Terminate bleed and feed after any SG Wide Range level is > 17%.

ANSWER: A

A. CORRECT. Since core exit TCs are lowering and there is concern with thermal stress to hot, dry SG, procedural direction is to initiate flow in controlled flow rate (50 X10^3 lbm/hr) to provide heat removal while minimizing stresses. This flow rate may be raised when SG WR level is > 17%. Feed and bleed is terminated at SG NR level > 14% NR.

B. INCORRECT. Since core exit TCs are lowering and there is concern with thermal stress to hot, dry SG, procedural direction is to initiate flow in controlled flow rate (50 X10^3 lbm/hr) to provide heat removal while minimizing stresses. This flow rate may be raised when SG WR level is > 17%. Feed and bleed is terminated at SG NR level > 14% NR. This answer would be correct if core exit TCs were Raising

C. INCORRECT. Feed flow should not be established to the faulted SG if intact one exists (procedural direction). The 17% WR Level requirement is to allow FW Flow to be raised. Feed and bleed is not terminated until SG NR level is > 14% NR.

D. INCORRECT. Feed Flow should only be established to ONE SG. The 17% WR Level requirement is to allow FW Flow to be raised. Feed and bleed is not terminated until SG NR level is > 14% NR.

OBJECTIVE: RO-C-EOP11-E18, RO-C-EOP11-E6
REFERENCE: OHP-4023-FR-H.1 Response to Loss of Secondary Heat Sink (Foldout Page item #4, Step 1 Caution, & Step 31)

Attachment Provided : 01-OHP-4023-FR-H.1 Response to Loss of Secondary Heat Sink pg. 4-36 and FOP.

KA - 000054 AA2.06
Loss of Main Feedwater (MFW)
Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW):
AFW adjustments needed to maintain proper T-ave. and S/G level
RO - 4.0 SRO - 4.3
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Question addresses Loss of Heat Sink and the operators ability to monitor the plant conditions and select the appropriate actions including AFW flow to control temperatures.

Original Question # - RO26-0077, RO26 AUDIT-62
Original Question KA - 00WE05 EA1.3

88. 088 002/SRO/OK/DIRECT/NRC EXAM 2004-110-1/000055 2.4.35/RO-C-EOP14-E9/NRCAUDIT07-0058/F/3

Unit 2 was stable at 100% power with the 2CD Emergency Diesel Generator tagged out for oil pump replacement. A loss of offsite power occurs and the 2AB EDG fails to start. It is estimated that it will take 1 hour to restore power.

Which ONE of the following denotes the required actions (if any) for Unit 2 Control Room cooling as per OHP-4023-ECA-0.0, Loss of All AC?

A. Unit 2 Control room cooling is NOT required since power will be restored in 1 hour.

B. Open doors to provide cooling to Vital cabinets within 30 minutes.

C. Crosstie ESW to Unit 1 and align ESW cooling water to the Control Room Air handling unit.

D. Start the Unit 2 North Control Room air handling unit since it is supplied by Unit 1 power.

ANSWER: B

A. INCORRECT	Analysis requires cabinet doors opened within 30 minutes.
B. CORRECT	Unit 2 fans have lost power. The cabinet doors must be opened within 30 minutes to ensure equipment (control and protection cards/circuits) temperatures stay within design limits.
C. INCORRECT	Aligning ESW will provide Backup cooling if the chiller is lost but without the fans this is NOT effective.
D. INCORRECT	Unit 2 North is supplied by Unit 2 power.

OBJECTIVE: RO-C-EOP14-E9
REFERENCE: 2-OHP-4023-ECA-0.0, Loss of All AC Power, Step 8; PSBD 12-OHP-4023-ECA-0.0, Background Document, Step 8.

KA - 000055 2.4.35
Loss of Offsite and Onsite Power (Station Blackout)
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 - Question requires knowledge of tasks required of AEOs outside the control room.

Original Question # - NRCAUDIT07-0058, NRC EXAM 2004-110-1
Original Question KA - 0000552435

89. 089 009/SRO/OK - ATTACHMENT/NEW/NEW/000067 2.4.50/RO-C-AS18-E6/RO-C-AS18-E6-1/H/3

Given the following conditions:

- Unit 1 is in Mode 3.
- Annunciator Panel 101 Drop 28, UNIT 1 PYRALARM ABN OR FIRE actuates.
- Annunciator Panel 101 Drop 79, 4KV AREA CO2 VOLT FAIL OR SYS ARMED/ABN actuates.
- The Turbine Tour AEO has reported that a fire does NOT exist.
- Investigation reveals that the ionization/smoke detectors for the 4 KV Switchgear Room AB (12-3 & 12-4) are inoperable.

Which ONE of the following describes the required actions, if any, per TRM 8.7.4 Fire Detection Instrumentation and TRM 8.7.8 Low Pressure CO_2 Systems?

Attachment Provided

A. No actions for TRM 8.7.4 nor TRM 8.7.8 are required since 3 infrared detectors are still operable and capable of automatic actuation.

B. Restore the ionization detectors to operable status within 14 days per TRM 8.7.4. No additional actions are required for TRM 8.7.8 since Manual Actuation is still available.

C. No actions for TRM 8.7.4 nor TRM 8.7.8 are required since only 1 Train of 4KV switchgear is required to be operable in Mode 3.

D. Establish hourly fire watch patrol per TRM 8.7.4 since 3 infrared detectors are still operable. No additional actions are required for TRM 8.7.8.

ANSWER D

A. INCORRECT	Combination of only Action A.1 for TRM 8.7.4.
B. INCORRECT	This would be actions for Function A of TRM 8.7.4 detectors since there are 5 within the 4KV Room. Also function A of TRM 8.7.8 if CO_2 was isolated for work in the area.
C. INCORRECT	2 Trains of 4KV switchgear are required in MODE 3
D. CORRECT	These are Detector Function B which requires action B.1 of TRM 8.7.4 and an Ionization and infrared are required for actuation so Actions B.1.1 AND B.1.2 of TRM 8.7.8 are required.

OBJECTIVE: RO-C-AS18-E6
REFERENCE: TRM 8.7.4, TRM 8.7.8, OHP-4024-101 Figures 8 & 16

Attachment Provided: U1 TRM 8.7.4 Fire Detection Instrumentation, TRM 8.7.8 Low Pressure CO_2 Systems, OHP-4024-101 Figures 8 & 16

KA - 000067 2.4.50
Plant Fire on Site
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 - Question tests operator knowledge of the operation of the 4 KV Switchgear Room CO2 system in response to alarms and required actions due to a partial system failure.

Original Question # - NEW 2012-089
Original Question KA - N/A

90. 090 004/SRO/OK/DIRECT/RO25 AUDIT-76/00WE02 2.4.18/RO-C-EOP09-E36/NRCAUDIT07-0993A/H/3

The first charging pump is shutdown during safety injection termination per OHP-4023-ES-1.1, Safety Injection Termination, following a Unit 2 LOCA.
The following parameter trends are noted:

- Subcooling is +41˚F and lowering.
- Pressurizer level is 25% and lowering.
- RCS pressure is 1680 psig and lowering.

Which ONE of the following describes parameter(s) that were affected by the charging pump shutdown and what procedure will be used to stabilize the plant?

Lower injection flow is causing ...

A. pressure to lower. Go to OHP-4023-ES-1.2, Post LOCA Cooldown and Depressurization.

B. pressure to lower. Go to OHP-4023-FR-C.2 Response to Degraded Core Cooling.

C. temperature to rise. Go to 2-OHP-4023-ES-1.2, Post LOCA Cooldown and Depressurization.

D. temperature to rise. Go to OHP-4023-FR-C.2 Response to Degraded Core Cooling.

ANSWER: A

A. CORRECT	After the first CCP is stopped, RCS pressure is checked to determine if 1 CCP will be sufficient to maintain control with the lower flow. If the leak is large enough to cause RCS pressure to lower, then a transition is made to ES-1.2 to address the leak.
B. INCORRECT	A transition is made to ES-1.2 since RCS Pressure is lowering. FR-C.2 Entry requires a Lower RVLIS Level (PZR is 25%). Plausible since subcooling is lowering.
C. INCORRECT	The primary concern is lowering pressure and level. The transition is made based on pressure.
D. INCORRECT	A transition is made to ES-1.2 since RCS Pressure is lowering. FR-C.2 Entry requires a Lower RVLIS Level (PZR is 25%). Plausible since subcooling is lowering.

OBJECTIVE: RO-C-EOP09-E40, RO-C-EOP09-E36
REFERENCE: 12-OHP-4023-ES-1.1 Background Step 5

KA - 00WE02 2.4.18
SI Termination
Emergency Procedures/Plan
Knowledge of the specific bases for EOPs.
RO - 3.3 SRO - 4.0
CFR - 41.10 / 43.1 / 45.13

KA Justification - Question requires operator knowledge of the basis behind the steps and the required transition contained within SI termination.

Original Question # - NRCAUDIT07-0993, CATAWBA2005, RO25 Audit-76
Original Question KA - 000009 2.4.6

91. 091 004/SRO/OK/DIRECT/RO23 AUDIT 096-2/00WE05 EA2.1/RO-C-EOP01-E23/NRCAUDIT07-0465/H/3

The Unit 1 operators are responding to a RED path on the Heat Sink Critical Safety Function (CSF). They are implementing OHP-4023-FR H.1, Response to Loss of Secondary Heat Sink, when they identify a RED path on the Core Cooling CSF.

Which ONE of the following describes the proper procedural implementation based on these conditions?

A. Perform OHP-4023-FR-H.1 until completion, then implement OHP-4023-FR-C.1, Response to Inadequate Core Cooling.

B. Suspend OHP-4023-FR-H.1, and immediately implement OHP-4023-FR-C.1, Response to Inadequate Core Cooling.

C. Concurrently implement OHP-4023-FR-C.1, Response to Inadequate Core Cooling, and OHP-4023-FR-H.1.

D. Perform only steps 1 through 17 of OHP-4023-FR-H.1, then implement OHP-4023-FR-C.1, Response to Inadequate Core Cooling.

ANSWER: B

A. INCORRECT	Core Cooling is a higher priority and should be implemented immediately. Plausible if operator confused on priority and/or rules to complete procedure prior to transition.
B. CORRECT	Core Cooling is a higher priority and should be implemented immediately.
C. INCORRECT	Heat sink is exited. Concurrent use is allowed with some AOPs.
D. INCORRECT	Plausible since the first 17 steps restore cooling.

OBJECTIVE: RO-C-EOP01-E23
REFERENCE: OHI-4023 Attachment 5, CSFSTs

KA - 00WE05 EA2.1
Loss of Secondary Heat Sink
Ability to determine and interpret the following as they apply to the Loss of Secondary Heat Sink:
Facility conditions and selection of appropriate procedures during abnormal and emergency operations
RO - 3.4 SRO - 4.4
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Question requires operator knowledge of CSF priorities and selection, hierarchy, and coordination of procedures addressing loss of heatsink.

Original Question # - NRCAUDIT07-0465, RO23 AUDIT 096-2
Original Question KA - G 2.4.22

92. 092 005/SRO/OK/DIRECT - REPEAT/NRC EXAM 2008-83/00WE07 EA2.2/RO-C-EOP10-E20/CM-40286/H/4

Conditions at 1400 hrs:

- Unit 2 safety injection occurred due to a LOCA
- Reactor coolant system pressure is 915 psig
- RCS T-Cold temperatures are 535˚F
- Core Exit Thermocouples (CETC) are 565˚F
- Crew has implemented OHP-4023-FR-C.2, Response to Degraded Core Cooling
- The SM has directed a cooldown to allow RHR to be placed in service

Assuming temperatures have been stable since 1300 hrs, what is the maximum cooldown rate allowed per OHP-4023-FR-C.2, and what is the lowest temperature that could be attained by 1600 hrs using that rate?

A. Cooldown at 60˚F/hr to 415˚F on RCS T-Colds

B. Cooldown at 60˚F/hr to 385˚F on CETCs

C. Cooldown at 100˚F/hr to 335˚F on RCS T-Colds

D. Cooldown at 100˚F/hr to 265˚F on CETCs

ANSWER: C

A. INCORRECT. 60˚F/hr Cooldown Rate - 60˚F/hr is plausible because it is the normal cooldown limit in NOPs.

B. INCORRECT. 60˚F/hr & CETC temps used. The reference temperature to use is the T-Cold, NOT Core exit temperature and this is 3 hours (1300-1600).

C. CORRECT. This procedure allows 100˚F/hr. (TS 3.4.3 limit). The reference temperature to use is the T-Cold over 2 hours (1400-1600).

D. INCORRECT. CETC temps used. The reference temperature to use is the T-Cold, NOT Core exit temperature and this is 3 hours (1300-1600).

OBJECTIVE: RO-C-EOP10-E20
REFERENCE: OHP-4023-FR-C.2 Step 14 and Background

KA - 00WE07 EA2.2
Saturated Core Cooling
Ability to determine and interpret the following as they apply to the Saturated Core Cooling:
Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments
RO - 3.3 SRO - 3.9
CFR - 41.7 / 41.10 / 43.5 / 45.13

KA Justification - Question tests knowledge of Procedural/TS limits and the indications used to adhere to those limits.

Original Question # - CM-40286, NRC EXAM 2008-83, RO25 Audit -27, CATAWBA2005
Original Question KA - 00WE07 EA2.2

93. 093 009/SRO/OK/DIRECT/NRC EXAM 2004-013/00WE12 2.1.32/RO-C-EOP07-E16/RO-C-EOP07-E16-1/F/3

Unit 2 has experienced a steamline break. None of the Steam Generator (SG) Stop Valves can be closed. OHP-4023-ECA-2.1, "Uncontrolled Depressurization of all Steam Generators," has been implemented.

Which ONE of the following statements is correct regarding Attachment A, Local SG Isolation?

A. Isolation of both steam supply lines to the TDAFP is allowed, regardless of the status of the other sources of feed flow to the SGs, since no secondary heat sink is intact.

B. Integrity must be restored to all SGs before the operator can transition to OHP-4023-E-2, Faulted Steam Generator Isolation, via the foldout page.

C. Valves are closed one loop at a time in order to restore integrity to at least one SG as early as possible.

D. The Operator is allowed to place the Stop Valve Dump Valve control switches to LOCKOUT only if the selected valve is NOT accessible for local isolation.

ANSWER: C

A. INCORRECT	Isolation of both TDAFP steam lines is NOT allowed if it is the only source of FW.
B. INCORRECT	If integrity is restored to any SG the transition is made.
C. CORRECT	Valves are closed one loop at a time in order to ensure a complete, local check of the valves for each SG to restore integrity to at least one SG as early as possible.
D. INCORRECT	This action may be required if the Dump valves require manual Closure to override standing Automatic closure signal.

OBJECTIVE: RO-C-EOP07-E16
REFERENCE: PSBD 12-OHP-4023-ECA-2.1 Background Document Step 1 Basis pg. 7 and Attachment A Basis pg. 82

KA - 00WE12 2.1.32
Uncontrolled Depressurization of all Steam Generators
Conduct of Operations
Ability to explain and apply system limits and precautions.
RO - 3.8 SRO - 4.0
CFR - 41.10 / 43.2 / 45.12

KA Justification - Questions requires operator knowledge of detailed implementation of attachments as well as precautions (TDAFP isolation) and limits placed on isolating at least 1 SG.

Original Question # - NRC Exam 2004-013, 01EOPC0716-1
Original Question KA - 000040 AK2.01

94. 094 002/SRO/OK/DIRECT/RO27AUDIT-94/194001 2.1.6/RO-C-ADM06-E5/RO27AUDIT-94/F/3

Given the following conditions:

- A tornado warning has been issued for the Cook Plant Area
- The SRO has entered 12-OHP-4022-001-010, Severe Weather

During a procedure implementation with offsite power still available, the SRO will direct the operators to ensure all EDGs are ______(1)________ to____________________(2)____________.

	(1)	(2)
A.	running in parallel with offsite power source	maintain positive control of electrical power to the plant
B.	running in parallel with offsite power source	allow isolation of safeguards bus from the RCP bus
C.	stopped and in standby	prevent a switchyard fault from damaging the EDG
D.	stopped and in standby	ensure that diesels are only started following verification of no damage

ANSWER: C

A - INCORRECT. EDGs are verified stopped and in standby if offsite power is available to ensure a switchyard fault does not damage the EDGs.

B - INCORRECT. EDGs are verified stopped and in standby if offsite power is available to ensure a switchyard fault does not damage the EDGs.

C - CORRECT. EDGs are verified stopped and in standby if offsite power is available to ensure a switchyard fault does not damage the EDGs.

D - INCORRECT. EDGs are verified stopped and in standby if offsite power is available to ensure a switchyard fault does not damage the EDGs.

OBJECTIVE: RO-C-ADM06-E5
REFERENCE: 12-OHP-4022-001-010, Step 4.

KA - 194001 2.1.6
Generic
Conduct of Operations
Ability to manage the control room crew during plant transients.
RO - 3.8 SRO - 4.8
CFR - 41.10 / 43.5 / 45.12 / 45.13

KA Justification - Matches the K/A because the applicant must determine what direction to give the operators concerning the EDGs during a tornado warning. SRO Only aspect is the usage of the given procedure to assess which action is required and the reason for this action.
Original Question # - RO27AUDIT-94
Original Question KA - 194001 2.1.8

95. 095 002/SRO/OK/DIRECT/NRC EXAM 2007-97/194001 2.1.35/RO-C-ADM13-E3/2007-0535/H/3

The plant is in MODE 6.
Fuel movement was suspended for repairs to the Spent Fuel Bridge Crane. Repairs to the Spent Fuel Bridge Crane are complete.

- Source Range Channel N31 is INOPERABLE
- Source Range Channels N32 and N23 are OPERABLE.
- The West RHR pump has just been placed in service due to the failure of the East RHR pump seal.
- The Reactor Cavity Water Level is 644' 6".

The refueling team has established communications with the control room, and has requested permission to move the next fuel bundle from the fuel building to the core.

Are administrative conditions met to recommence fuel movement?

A. Yes, but only if the Reactor Cavity Water Level is raised to greater than 644' 9"

B. No, the East RHR pump must be restored to OPERABLE.

C. No, Source Range Channel N31 must be restored to OPERABLE.

D. Yes, provided that the Audible count rate circuit is selected to N32.

ANSWER: D

A. INCORRECT	Level is only required to be 644' 1.5". 644' 9" is nominal.
B. INCORRECT	Only 1 RHR is required to be operable with > 23'.
C. INCORRECT	The gamma Metrics may be used for the second source range channel.
D. CORRECT	For refueling to begin, 2 SR channels are required, one with an audible count rate indication. This comes only from N31 or N32. The gamma metrics (N23) may be used for the other channel. One RHR pump must be operating and level must be > 23' or 644' 1.5". The conditions are met for refueling provided that the audible count rate is selected to N32.

OBJECTIVE: RO-C-ADM13-E3
REFERENCE: 1-OHP-4030-127-037, Refueling Surveillance, Data Sheet 2 & 3

KA - 194001 2.1.35
Generic
Conduct of Operations
Knowledge of the fuel-handling responsibilities of SROs.
RO - 2.2 SRO - 3.9
CFR - 41.10 / 43.7

KA Justification - Question requires knowledge of SRO Administrative responsibilities and verifications required prior to movement of fuel.

Original Question # - NRC EXAM 2007-97, 2007-0535
Original Question KA - Generic 2.2.26

96. 096 023/SRO/OK - ATTACHMENT/DIRECT/RO27 AUDIT -90/194001 2.2.40/RO-C-TS01-E11/RO27AUDIT-90/H/3

Given the following conditions:

- At 1800 on May 13, SR 3.8.9.1 (verifies that DC Distribution system breaker lineup is correct at a specified frequency of 7 days) was performed satisfactorily.

- On May 23 at 1700, it is discovered by the Shift Manager that SR 3.8.9.1 was last performed on May 13.

Which ONE of the following is correct regarding the status of the DC Distribution and requirements for SR 3.8.9.1 performance?

Attachment Provided

The affected DC Distribution is ____

A. INOPERABLE because the delay period beyond the maximum extension time has been exceeded. SR 3.8.9.1 must be completed satisfactorily by May 23 at 1900.

B. INOPERABLE if SR 3.8.9.1 is not completed satisfactorily by May 23 at 1900.

C. OPERABLE. If a risk evaluation is performed to determine risk impact, then SR 3.8.9.1 completion can be extended at the latest, to May 27 at 1800.

D. OPERABLE. If a risk evaluation is performed to determine risk impact, then SR 3.8.9.1 completion can be extended at the latest, to May 30 at 1700.

ANSWER: D

A - INCORRECT. The SRO must consider the original specified frequency of 1.25 X 7 days = 8 days and 18 hours. That frequency has been exceeded, and the applicant incorrectly believes the batteries are now inoperable. Applicant may also mistakenly apply the LCO 3.8.4, Condition A Completion Time of 2 hours to the time of discovery and believe that the surveillance must be performed no later than time of discovery plus the 2 hours; i.e., discovered late on May 23 at 1700 plus two hours = May 23 at 1900.

B - INCORRECT. While it is true that the batteries can be considered operable (as explained in 'd' below), the SR completion can go past 24 hours providing a risk assessment is completed.

C - INCORRECT. While it is true that the batteries can be considered operable (as explained in 'd' below), and that a risk evaluation can allow a completion time extension of up to the specified frequency (7 days), the given completion is incorrect. Applicant incorrectly calculates the 7 day frequency from the original 7 day frequency expiration; i.e., May 13 + 7 days = May 20 + 7 days = May 27. The correct time is from the time of discovery, plus 7 days (specified frequency), per SR 3.0.3.

D - CORRECT. SR 3.0.3, 'Surveillance Requirement Applicability,' permits delaying the requirement to declare the LCO not met, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. However, if the surveillance is delayed greater than the 24 hour period, a risk impact must be performed to determine the scope of any risk. Since the SR was last performed on May 13 at 1800, it should have been performed by May 22 at 1200 (specified frequency = 7 days + .25 maximum extension time = 8 days and 18 hours). The discovery (29 hours later) that it had not been performed does not exceed the specified frequency of the surveillance if a risk assessment is performed. The required completion time is calculated by adding the specified frequency (7 days) to the time of discovery; i.e., May 23 at 1700 + 7 days = May 30 at 1700.

OBJECTIVE: RO-C-TS01-E11
REFERENCE: TS 3.8.9, TS SR 3.0.3

ATTACHMENT Provided: Unit 1 TS Section 3.0, and 3.8.9

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 - Applicant is presented with a plant condition involving an overdue surveillance and how that affects operability of the equipment. The SRO Only aspect is in understanding and applying Tech. Spec. Completion Time Extension rules.

Original Question # - RO27 Audit -90, WattsBarMay2009-90
Original Question KA - 063000 2.2.37

97. 097 006/SRO/OK/DIRECT/RO26 AUDIT-81/194001 2.3.6/RO-C-02800-E12/RO26-0026/H/3

Given the following plant conditions:

- Unit 2 is in Mode 5
- Actions are being taken to start the Containment Purge system in the Clean-Up Mode.
- Due to an inadvertent overload trip of the Containment Purge supply fan, there has been a 30 hour delay since Radiation Protection (RP) approval to start the purge.

An operator is ready to start aligning purge and asked for Unit Supervisor permission to continue. Which ONE of the following describes the appropriate response by the Unit Supervisor?

A. Purge may be initiated since RP approval was within 48 hours of the start of the purge operation.

B. Purge may be initiated as long as Plant Manager notification made within 24 hours of purge initiation.

C. Purge can NOT be placed in service due to exceeding 24 hours from the time of initial RP approval.

D. Purge can NOT be placed in service until a second air sample is obtained from the containment atmosphere.

ANSWER: C

A. INCORRECT. Purge must be initiated within 24 hours of RP approval.
B. INCORRECT. Plant Manager approval is required prior to purge operation for purges that are not approved by RP.
C. CORRECT. Purge must be initiated within 24 hours of RP approval.
D. INCORRECT. RP approval is required within 24 hours of the purge initiation.

OBJECTIVE: RO-C-02800-E12, SR-O-0001/Task# 0280350103
REFERENCE: 2-OHP-4021-028-005, Attachment 1, Step 3.1.

KA - 194001 2.3.6
Generic
Radiation Control
Ability to approve release permits.
RO - 2.0 SRO - 3.8
CFR - 41.13 / 43.4 / 45.10

K/A Justification - Question tests the ability to determine if the requirements for performing a containment purge (radiation release) have been satisfied.

Original Question # - RO26-0026, RO26 AUDIT-81
Original Question KA - 022000 2.3.11

98. 098 004/SRO/OK/NEW/NEW/194001 2.3.14/RO-C-00200-E10/RO-C-00200-E10-1/F/3

Given the following plant conditions:

- A rapid load reduction from 100% power to 60% power was performed on Unit 1 approximately 3 hours ago.
- Chemistry confirms that RCS I-131 activity exceeds Technical Specification limit of acceptable operation.
- The US directs a plant shutdown to be performed .

Which ONE of the following post shutdown actions is subsequently performed to limit the release of activity?

A. All Steam Generator Stop Valves are closed.

B. Reactor Coolant System temperature is reduced below 500°F.

C. All Steam Generator PORV setpoints are raised.

D. Reactor Coolant System letdown flow is diverted to the CVCS Hold Up Tank (HUT).

ANSWER: B

A. INCORRECT. Closing SGSVs does not prevent rad release from SG PORVs.
B. CORRECT. Reduce temp IAW TS 3.4.16.
C. INCORRECT. Would not stop a release from SG Safety Valves.
D. INCORRECT. Letdown Flow is typically maximized and the Cation Demin may be placed in service. Plausible since Letdown is diverted to prevent contaminating the VCT following Demin excursions.
OBJECTIVE: RO-C-00200-E10
REFERENCE: TS & Bases 3.4.16 RCS Specific Activity.

KA - 194001 2.3.14
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 - Question requires operator knowledge of the radiation hazards and mitigating actions required for high RCS activity.

Original Question # - NEW 2012-98
Original Question KA - N/A

99. 099 002/SRO/OK/DIRECT/NRC EXAM 2004-130-3/194001 2.4.8/RO-C-EOP01-E25/RO26-0033/F/3

Given the following plant conditions:

- Unit 2 was operating at 100% power with the West CCW pump Tagged Out.
- The East CCW pump tripped and could NOT be restarted, resulting in the loss of CCW.

Which ONE of the following describes the correct Operator response?

Immediately trip the Reactor and implement ...

A. OHP-4023-E-0, Reactor Trip or Safety Injection.
Trip the RCPs after reactor trip has been verified.
OHP-4022-016-004, Loss of CCW, may be performed concurrently after the immediate actions are complete.

B. OHP-4023-E-0, Reactor Trip or Safety Injection.
Trip the RCPs after the immediate actions are complete.
OHP-4022-016-004, Loss of CCW, is NOT needed since the EOP network addresses a loss of CCW.

C. OHP-4023-E-0, Reactor Trip or Safety Injection.
Trip the RCPs after the immediate actions are complete.
Steps from OHP-4022-016-004, Loss of CCW, may NOT be performed until completion of OHP-4023-ES-0.1, Reactor Trip Response.

D. OHP-4022-016-004, Loss of CCW, until restoration of CCW from any source.
Trip the RCPs after reactor trip has been verified.
Perform OHP-4023-E-0, Reactor Trip or Safety Injection steps as time allows.

ANSWER: A

A. CORRECT. OHI-4023, Abnormal/Emergency Procedure User's Guide allows Abnormal Procedures to be implemented concurrently with Emergency Procedures after the immediate actions are complete. The RCPs are tripped after the reactor trip has been verified per the Note in OHP-4022-016-004.

B. INCORRECT. Performance of OHP-4023-E-0 is required upon the reactor trip, but the operators must continue to perform OHP-4022-016-004 to address the loss of CCW.

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

D. INCORRECT. The Unit Supervisor should direct action of OHP-4023-E-0, first, NOT as time allows. OHP-4023-E-0 actions take priority over OHP-4022-016-004.

OBJECTIVE: RO-C-EOP01-E25
REFERENCE: OHI-4023 Abnormal/Emergency Procedure User's section 4.6.9; 2-OHP-4022-016-004 step 2 & Note

KA - 194001 2.4.8
Generic
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 - Required the knowledge of the strategies for implementing the EOPs coincident with a Loss of CCW AOP.

Original Question # RO26 AUDIT-25, RO26-0033, NRCAUDIT07-0108, NRC EXAM 2004-130-3
Original Question KA - 000026 2.4.6

100. 100 015/SRO/OK/DIRECT/RO27 AUDIT -97/194001 2.4.29/ST-C-EP01-E2/RO27AUDIT-97/H/3

Given the following conditions occur on Unit 1:

- A Loss of Coolant Accident (LOCA) has occurred.
- A Site Area Emergency (SAE) has been declared.
- The Technical Support Center (TSC) has been activated.
- The Operation Support Center (OSC) has been activated.
- It is necessary to enter Unit 1 Safety Injection Pump Room to prevent core damage.
- Projected dose rate in the pump room is 1.16E+5 mR/hr.
- Duration of the exposure is expected to be 4 minutes.

Which ONE of the following is the lowest authority (by title) who can authorize the exposure as specified in RMT-2080-TSC-001, Activation and Operation of the TSC, Attachment 13?

A. Radiation Protection Manager

B. Site Emergency Coordinator

C. OSC Manager

D. Operations Shift Manager

ANSWER: B

A - INCORRECT. RP Manager is not authorized to approve emergency dose exposures in accordance with RMT-2080-TSC-001, but is plausible since the Radiation Protection Manager has authority over many functions of the Radiation Control group.

B - CORRECT. In accordance with RMT-2080-TSC-001, the site SEC is the only individual who is authorized to approve emergency dose exposures and the authority cannot be delegated to anyone else. Dose is above the 5 Rem annual whole body limit.

C - INCORRECT. OSC Manager is not authorized to approve emergency dose exposures in accordance with RMT-2080-TSC-001, but is plausible since the OSC Manager dispatches response teams.

D - INCORRECT. Operations Shift Manager is no longer the SEC once the TSC is activated. Therefore he is not authorized to approve emergency dose exposures in accordance with RMT-2080-TSC-001, but is plausible since the Operations Shift Manager has authority over many functions relating to plant and site operations.

OBJECTIVE: ST-C-EP01-E2
REFERENCE: ST-C-EP01, RMT-2080-TSC-001, Attachment 13

KA - 194001 2.4.29
Generic
Emergency Procedures/Plan
Knowledge of the emergency plan.
RO - 3.1 SRO - 4.4
CFR - 41.10 / 43.5 / 45.11

KA Justification - Applicant must evaluate a condition involving a proposed emergency radiation exposure and apply knowledge of exposure limits to determine the category of exposure and who must authorize it in accordance with the emergency plan including knowledge of the SRO duties of SEC and the transfer to TSC.

Original Question # - RO27 Audit -97, WattsBarMay2009
Original Question KA - 194001 2.3.4