# EXAMINATION

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

**1** **ID: 1685138** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Reactor Trip due to low condenser vacuum
- Condenser vacuum is 22 in Hg mercury and slowly lowering
- OTSG 'A' is at 98% in the operating range and slowly rising
- OTSG 'B' is at 25" in the startup range and steady

What actions will the operating crew take to stabilize the plant?

A. 1) Trip 'A' Main Feed Pump
   2) Stabilize OTSG pressure with the MS-V-3's (Turbine Bypass Valves)

B. 1) Trip both Main Feed Pumps
   2) Stabilize OTSG pressure with the MS-V-4's (Atmospheric Dump Valves)

C. 1) Trip 'A' Main Feed Pump
   2) Stabilize OTSG pressure with the MS-V-4's (Atmospheric Dump Valves)

D. 1) Trip both Main Feed Pumps
   2) Stabilize OTSG pressure with the MS-V-3's (Turbine Bypass Valves)

**2** **ID: 1737142** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- A Small Break LOCA has occurred inside containment.
- The leak is estimated at 25 gpm.
- Reactor Building pressure is 0.5 psig and rising at approximately 0.1 psig every two hours.
- Reactor Building temperature above 320' elevation is 120°F and rising at approximately 2°F every hour.

OTSG Startup level indicates __(1)__ than actual based on elevated Reactor Building __(2)__.

A. (1) lower
(2) pressure

B. (1) higher
(2) pressure

C. (1) lower
(2) temperature

D. (1) higher
(2) temperature

**3** **ID: 1685156** **Points: 1.00**

Plant Conditions:

- Reactor is a 70% power
- All ICS stations are in HAND

EVENT:

- RC-P-1B trips

Tcold on the 'A' RCS loop will _______ and the operator will have to _______ 'A' feedwater flow to maintain a $0^0$F $\Delta$Tc.

A. rise
raise

B. rise
lower

C. lower
lower

D. lower
raise

**4** **ID: 1685199** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- MU-P-1B trips on overload.
- MAP G-2-5 PZR LEVEL HI-LO annunciator has alarmed.
- Pressurizer level is 196 inches and slowly lowering.

Which one of the following identifies actions that the operator must take while restoring a Makeup Pump to operation?

A. De-energize all Pzr Heaters; AND
CLOSE MU-V-33A-D, Control Bleed Off Valves.

B. CLOSE MU-V-3, Letdown Isolation Valve; AND
CLOSE MU-V-33A-D, Control Bleed Off Valves.

C. De-energize all Pzr Heaters; AND
Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to HAND and CLOSE the valves.

D. CLOSE MU-V-3, Letdown Isolation Valve; AND
Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to HAND and CLOSE the valves.

**5** **ID: 1737146** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- All available Nuclear River water pumps trip, and cannot be restarted.
- OP-TM-AOP-031, LOSS OF NUCLEAR SERVICES COMPONENT COOLING is entered.

Which one of the following describes the strategy and basis for Reactor Coolant Pump (RCP) operation?

A. The RCPs must be secured due to the loss of labyrinth cooling to each RCP.

B. The RCPs must be secured to eliminate heat load from the NSCCW System.

C. The RCPs may remain in operation provided seal injection flow is greater than 22 gpm.

D. The RCPs may remain in operation provided the SR-NR cross tie is performed prior to reaching the HI-2 alarm setpoint for the RCPs.

**6** **ID: 1685231** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- The PORV fails partially open.
- Efforts to close the PORV Block valve have failed.

Current Plant Conditions:

- The plant is being shutdown.
- RCS Pressure is 2050 psig and slowly lowering.
- RCS Subcooling margin is 36°F.
- Pressurizer level is 250 inches and slowly rising.

In accordance with OP-TM-AOP-043, LOSS OF PRESSURIZER, assuming the plant does NOT go solid, what action must be taken regarding operation of MU-V-17, Pressurizer Level Control Valve?

Place MU-V-17 in hand and adjust makeup flow to maintain ___(1)

A. > 40°F SCM by squeezing the pressurizer bubble.

B. > 150 inches in the pressurizer to ensure adequate inventory on the subsequent RCS cooldown.

C. > 70°F SCM by squeezing the pressurizer bubble.

D. > 220 inches in the pressurizer to ensure adequate inventory on the subsequent RCS cooldown.

**7** **ID: 1685342** **Points: 1.00**

Event:

- Loss of offsite power.
- Emergency Diesel Generator, EG-Y-1A fails to start.
- Emergency Diesel Generator, EG-Y-1B trips after starting.

Based on these conditions, which of the following control room extension controls or instruments remain operable?

A. RM-A-2 Sample Pump.

B. Motor Driven Fire Pump FS-P-2.

C. 1SB-D2, 1B Auxiliary Transformer feeder breaker to 1D 4KV Switchgear.

D. RR-PI-225, RB Emergency Cooling Outlet Cooler 1B pressure instrument.

**8** **ID: 1685400** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- Station Blackout Diesel Generator, EG-Y-4 is unavailable due to an outage.

EVENT:

- Loss of Offsite power.
- Emergency Diesel Generator, EG-Y-1B has started and is powering 'E' 4kV Bus.
- Emergency Diesel Generator, EG-Y-1A has tripped on overspeed.

What action per OP-TM-AOP-020, Loss of Station Power, will be taken to prevent the loss of plant control and provide additional time to restore AC power?

A. Secure Emergency DC Seal Oil Pump GN-P-2.

B. Initiate OP-TM-732-901, Energize 1P 480V Bus using the ES Bus Cross Tie.

C. Initiate OP-TM-732-902, Energize 1S 480V Bus using the ES Bus Cross Tie.

D. Ensure Turning Gear Oil Pump LO-P-5 is running, then secure Emergency Bearing Oil Pump LO-P-6.

**9** **ID: 1685632** **Points: 1.00**

Plant Conditions:

- Plant Cooldown is in progress per 1102-11, Plant Cooldown.
- Current Cooldown rate is 85°F/HR.
- Current RCS temperature is 359°F and RCS Pressure is 600 psig.

EVENT:

- Channel 1 of B train ES indicates actuated.
- NI-5 loses indication on console center.
- MU-V-2A/2B Letdown Cooler Outlet Valves close.

(1) Which one of the following describes the action required if RCS temperature lowers below 313°F?
(2) What is the reason for the action taken?

A. (1) Open RC-RV-2 (PORV) until RCS pressure is below 540 psig.
(2) The NDTT interlock has lost power.

B. (1) Open RC-V-1 (RCS Spray Valve) until RCS pressure is below 540 psig.
(2) The PORV and the NDTT interlock have lost power.

C. (1) Open RC-RV-2 (PORV) until RCS pressure is below 540 psig.
(2) RC-V-1 and the NDTT Interlock have lost power.

D. (1) Open RC-V-1 (RCS Spray Valve) until RCS pressure is below 540 psig.
(2) The NDTT interlock has lost power.

**10** **ID: 1699640** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- Operating crew is performing Emergency Diesel Generator, EG-Y-1A, 24 hour endurance run in accordance with 1107-3.
- Currently EG-Y-1A is operating in parallel with the grid.

EVENT:

The following alarms are received simultaneously:

- Battery 1A Discharging, A-1-7.
- Battery Charger 1A/1C/1E Trouble, A-2-7.
- Inverter 1A/1C/1E System Trouble, A-3-7.
- CRDM Breaker Test Trouble, PRF 1-1-1.
- 230 KVolt Substation Trouble, NN-3-1.
- 7 KVolt Bus Trouble, AA-3-2.
- 4 KVolt BOP Bus Trouble, AA-3-3.
- 480 Volt BOP Bus Trouble, AA-3-5.

In order to protect EG-Y-1A the operating crew must:

G1-02 = EG-Y-1A Diesel output breaker to 1D 4160V Bus
EG-V-15A = Air Start Header Isolation Valve to EG-Y-15A

A. Close EG-V-15A, only.

B. Close EG-V-15A and trip EG-Y-1A fuel rack.

C. Open breaker G1-02 from the control room and close EG-V-15A.

D. Open breaker G1-02 using the local OPEN pushbutton and close EG-V-15A.

# EXAMINATION

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**11** **ID: 1699734** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- NR-P-1C is out of service for maintenance.
- NR-P-1A is ES selected and running on 1R 480v bus.
- NR-P-1B is ES selected and running on 1T 480v bus.

EVENT:

- Loss of Offsite Power occurs.
- RCS pressure lowered to 1575 psig and is now rising slowly.
- RB pressure is 1.5 psig and relatively steady.

EVENT + 10 seconds:

- The Diesel Generators have started and are powering the ES Buses.

Which one of the following identifies the Nuclear River Water pump(s) that are running, if any, 1 minute after the re-energization of the ES Buses?

A. NR-P-1A ONLY.

B. NR-P-1B ONLY.

C. NR-P-1A and NR-P-1B.

D. No NR pump is running.

**12** **ID: 1700681** **Points: 1.00**

<<REFERENCE PROVIDED

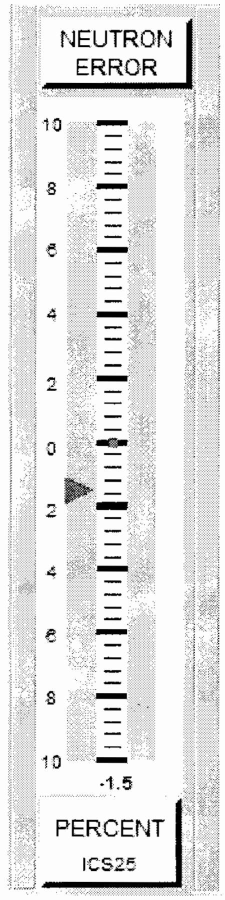
Plant Conditions

- Reactor power is 75% with ICS in auto.

EVENT:

- NI-5 power starts to slowly lower causing a SASS Mismatch.
- FW-V-17A, Main Feedwater Control Valve, starts to fail closed, causing a Feedwater to Reactor Cross Limit.
- Neutron Error is at -1.5% on console center.

Based on the above plant conditions, control rods will __(1)__ due to __(2)__.




A. (1) insert
(2) NI-5 failing

B. (1) insert
(2) Feedwater to Reactor Cross Limits

C. (1) withdraw
(2) NI-5 failing

D. (1) withdraw
(2) Feedwater to Reactor Cross Limits

**13** **ID: 1741874** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- Pressurizer Level LT-1 is selected as the controlling channel.

EVENT:

- A leak develops on the reference leg of LT-1 causing pressurizer level to change 100 inches over the next 5 seconds.

Based on these conditions, indicated (LT-1) Pressurizer level has ___(1)___ and assuming no operator action ___(2)___.

A. (1) risen
(2) MU-V-17, Pressurizer Level Control Valve, will close

B. (1) risen
(2) the controlling Pressurizer Level Transmitter will SASS from LT-1 to LT-3

C. (1) lowered
(2) MU-V-17, Pressurizer Level Control Valve, will open

D. (1) lowered
(2) the controlling Pressurizer Level Transmitter will SASS from LT-1 to LT-3

**14** **ID: 1700748** **Points: 1.00**

Which one of the following requires a reactor startup to be aborted?

A. No source range nuclear instrumentation when both intermediate channels reading ~ $5x10^{-11}$ amps.

B. No intermediate range nuclear instrument channels with 2 of 4 power range channels greater than 10% full power.

C. Expected criticality 30% lower than the Estimated Critical Position and on the next pull.

D. Most conservative valid wide range cold leg temperature indication shows RCS temperature less than 530F.

**15** **ID: 1737531** **Points: 1.00**

When required to verify a Radiation Monitoring System Setpoint, which is the PREFERRED METHOD for obtaining a Radiation Monitor's actual setpoints?

A. Contacting the I&C Department to verify the setpoint.

B. Performing a Source Check IAW 1301-4.1, Weekly Surveillance Checks.

C. Reading Alarm setpoints from the meter face by pressing the alarm pushbutton.

D. Refer to the setpoints section of alarm response OP-TM-MAP-C0101, RADIATION LEVEL HI.

**16** **ID: 1700780** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto..

EVENT:

- MAP C-1-1, RADIATION LEVEL HI alarm illuminates.
- RM-L-1-LO (RCS Letdown Rad Monitor) indicates a HIGH ALARM.
- RM-L-1-HI (RCS Letdown Rad Monitor) indicates an ALERT ALARM.

Based on the given conditions, letdown flow __(1)__ isolated, because __(2)__.

A. (1) IS
(2) RM-L-1-LO HIGH alarm closes MU-V-1A and MU-V1B, Letdown Cooler Inlet valves

B. (1) IS
(2) RM-L-1-LO HIGH alarm closes MU-V-2A and MU-V-2B, Letdown Cooler Outlet valves

C. (1) IS NOT
(2) both RM-L-1-LO AND RM-L-1-HI have to be in HIGH Alarm

D. (1) IS NOT
(2) RM-L1-LO HIGH alarm does NOT close MU-V-2A and MU-V-2B

**17** **ID: 1720243** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- The ICS Selected Feedwater Loop Temperature transmitter slowly fails high.

The Total Feedwater demand signal will ___(1)___ due to ___(2)___.

A. (1) rise
(2) BTU limits

B. (1) rise
(2) Feedwater Temperature Modification

C. (1) lower
(2) BTU limits

D. (1) lower
(2) Feedwater Temperature Modification

**18** **ID: 1737211** **Points: 1.00**

To avoid damage due to gas in the Makeup Pumps, __(1)__ must be closed if Makeup Tank level lowers to less than __(2)__ inches with suction from LPI or the BWST.

A. (1) MU-V-18, RCS Makeup Isolation Valve
(2) 18 inches

B. (1) MU-V-12, Makeup Tank Outlet Isolation Valve
(2) 18 inches

C. (1) MU-V-18, RCS Makeup Isolation Valve
(2) 40 inches

D. (1) MU-V-12, Makeup Tank Outlet Isolation Valve
(2) 40 inches

**19** **ID: 1737168** **Points: 1.00**

REFERENCE PROVIDED

Plant Conditions:

- Plant Cooldown in Progress
- All 4 Reactor Coolant Pumps are OFF
- DH-P-1A is Operating.
- RCS Temperature = 170F
- RCS Pressure = 250 psig

Curve __(1)__ prevents exceeding LTOP limits. The temperature instrument monitored to prevent exceeding these limits is __(2)__.

A. (1) Curve A
(2) DH6-TI-1 Decay Heat Suction Temperature

B. (1) Curve B
(2) DH6-TI-1 Decay Heat Suction Temperature

C. (1) Curve A
(2) DH2-TI-1 Decay Heat Cooler Outlet Temperature

D. (1) Curve B
(2) DH2-TI-1 Decay Heat Cooler Outlet Temperature

**20** **ID: 1700999** **Points: 1.00**

Plant conditions:

- Plant Cooldown is in progress in accordance with 1102-11.
- RCS pressure is 60 psig and steady prior to venting.
- OP-TM-220-552, VENTING THE PRESSURIZER TO THE RCDT (WDL-T-3), has been initiated to lower pressurizer pressure.

To prevent large pressure transients during the Pressurizer venting, the operator must ________

A. open RC-V-18, Manual Pzr Vent to RCDT.

B. throttle open RC-V-28, Pressurizer Vent Valve, for 1 second.

C. throttle open RC-V-1, Spray Valve, to help lower Pressurizer pressure.

D. ensure RCDT cooling pump (WDL-P-8) is in AUTO with both ICCW Pumps running.

**21** **ID: 1701024** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- EFW Area Cooling Fan, AH-E-24A, is operating.
- EFW Area Cooling Fan, AH-E-24B, is in standby.
- Emergency Feedwater Pump, EF-P-2A is running for an IST.

EVENT:

- Nuclear Service Closed Cooling Water Inlet Temperature Controller to AH-E-24A Coolers, TC-857, Fails LOW.

As a result of this controller failure, identify the resulting effect(s):

AH-E-24A trips due to high ___(1)___ temperature, and AH-E-24B ___(2)___.

A. (1) outlet
(2) automatically starts

B. (1) outlet
(2) can be started manually

C. (1) motor
(2) automatically starts

D. (1) motor
(2) can be started manually

**22** **ID: 1701070** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Pressurizer heater pressure control SETPOINT (RC3-PIC) signal fails to zero (0) psig.

What action(s) are required?

A. Raise pressurizer level to maintain pressure until heaters are restored.

B. Place the Pressurizer Spray Valve, RC-V-1 in Manual and CLOSE RC-V-1.

C. Verify all pressurizer heaters are deenergized OR manually deenergize heaters.

D. Adjust Heater Banks 1, 2 & 3 using RC3PIC, Pressurizer Pressure control in hand.

# EXAMINATION

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**23** **ID: 1701075** **Points: 1.00**

Initial plant conditions:

- Reactor trip due to LOCA.
- Containment Building pressure 5 psig, steady.
- Manual ESAS actuation signals were NOT initiated.
- Automatic ES Actuation status:

| ES Actuation | Train A | Train B |
|---|---|---|
| 1600# | Actuated (NOT Bypassed) | Bypassed |
| 500# | Not Actuated | Not Actuated |
| 4# | Defeated | Actuated (NOT Defeated) |

EVENT:

- Containment Building pressure rises rapidly to 35 psig.

Based on these conditions, identify the ONE statement below that describes the response of the Reactor Building Spray System

A. Only BS-P-1A starts.

B. Only BS-P-1B starts.

C. Both BS-P-1A and BS-P-1B start.

D. Neither BS-P-1A nor BS-P-1B starts.

**24** **ID: 1701092** **Points: 1.00**

REFERENCE PROVIDED

Plant Conditions:

- 100% power with ICS in full auto.
- RB Cooling Fans AH-E-1A/B/C are all operating in FAST SPEED.

EVENT:

- THERMAL OVERLOAD (49F) actuates.
- Thermal overload condition has NOT been reset at AH-E-1A breaker.

SUBSEQUENT EVENT:

- Main Steam System leak inside the Containment Building.
- RB Temperature Elevation > 320 is at 132F and slowly rising.
- NN-2-7 RB AIR TEMP HI is in Alarm.
- RB Pressure 2.3 psig.

Based on these conditions, AH-E-1A is __(1)__. In accordance with OP-TM-534-901, RB EMERGENCY COOLING OPERATIONS, AH-E-1A must be started in /shifted to __(2)__ speed.

A. (1) tripped
(2) slow

B. (1) running
(2) slow

C. (1) tripped
(2) fast

D. (1) running
(2) fast

# EXAMINATION

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**25** **ID: 1718554** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- OP-TM-621-251 SASS Logic Test in Progress.
  - Test 'A' on NI-5 / NI-6 Rx Pwr was successful, however the SASS channel FAILED to reset.
- NI-5 is selected for control.

EVENT:

- NI-5 upper chamber power supply fails to zero volts.

The crew __(1)__, because __(2)__.

A. (1) must immediately enter OP-TM-AOP-070, Primary to Secondary Heat Transfer Upset
(2) reactor power would be lowering

B. (1) must immediately enter OP-TM-AOP-070, Primary to Secondary Heat Transfer Upset
(2) reactor power would be rising

C. (1) must immediately enter OP-TM-EOP-001, Reactor Trip
(2) the NI-5 failure would cause a reactor trip

D. (1) would have NO required actions
(2) the SASS would automatically transfer to NI-6 for reactor control

**26** **ID: 1720398** **Points: 1.00**

Which one of the following describes the operation of the Main Feedwater Pump Monitoring circuit that provides Emergency Feedwater with an input from the Heat Sink Protection System?

EACH PUMP uses:

A. Two pressure bistables to sense Main Feedwater Pump discharge pressure.

B. Three pressure bistables to sense Main Feedwater Pump discharge pressure.

C. Two pressure bistables to sense hydraulic oil pressure at the Main Feedwater Pump turbine stop valves.

D. Three pressure bistables to sense hydraulic oil pressure at the Main Feedwater Pump turbine stop valves.

**27** **ID: 1737222** **Points: 1.00**

Plant conditions:

- Reactor critical at $10^{-8}$ amps.
- Crew is performing a middle of cycle startup from a 3 week forced outage.
- ICS in HAND, EXCEPT for the Main & Startup Feedwater Valves and Turbine Bypass Valves.
- Power escalation is on hold; will recommence in one hour.

EVENT:

- MS-V-3D, Turbine Bypass Valve fails OPEN

Assume NO Operator actions.

Which statement below describes the plant response to this failure?

A. RCS pressure lowers and the Reactor trips on low pressure.

B. Reactor power increases to the 'Point of Adding Heat' and stabilizes at ~1%.

C. Reactor power increases above the 'Point of Adding Heat' and then stabilizes at ~3%.

D. Reactor power increases to the 'Point of Adding Heat' and then lowers to $10^{-8}$ Amps in the Intermediate Range.

**28** **ID: 1717325** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- EF-P-2B, Motor Driven Emergency Feedwater Pump, is out of service for maintenance.

EVENT:

- All four Reactor Coolant Pumps trip.
- EF-P-1, Steam Driven Emergency Feedwater Pump, tripped on overspeed.
- 1D 4160V ES bus tripped on overcurrent.
- Subcooling margin is 35F and slowly lowering.
- Incore temperatures are rising.
- 'A' OTSG Level is at 36% Operating Range
- 'B' OTSG Level is at 30% Operating Range

Based on these plant conditions, the operator is required to feed the OTSG's with __(1)__ at a rate of __(2)__ to promote __(3)__.

A. (1) Main Feedwater
(2) >1 Mlbm/hr
(3) Natural Circulation

B. (1) Main Feedwater
(2) >1 Mlbm/hr
(3) Boiler Condenser Cooling

C. (1) Emergency Feedwater
(2) > 215 GPM /OTSGs
(3) Boiler Condenser Cooling

D. (1) Emergency Feedwater
(2) < 515 GPM
(3) Natural Circulation

**29** **ID: 1717742** **Points: 1.00**

In-accordance-with 1107-2C Vital DC Electrical System, which of the following local indications will tell an operator that a battery charger is in the EQUALIZE mode?

A. The AC Pilot Light is lit.

B. The current flow is at least 50 amps.

C. The output voltage is approximately 135 volts.

D. Both the AC & DC input and output breakers are closed.

**30** **ID: 1717423** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- 1M DC Bus is Powered from 1A DC Distribution Panel.

EVENT:

- Steam leak in the Reactor Building.
- The Reactor is tripped.
- RB Pressure increases to 4.2 psig.

10 Minutes Later:

- Loss of Offsite Power

Based on the above plant conditions:

__(1)__transfer capability of 1M DC Bus to its alternate power supply is blocked.

This transfer capability is blocked due to the __(2)__ signal.

A. (1) Only the auto, (2) undervoltage

B. (1) Only the auto, (2) Engineered Safeguards

C. (1) Both the auto and manual, (2) under voltage

D. (1) Both the auto and manual, (2) Engineered Safeguards

**31** **ID: 1717750** **Points: 1.00**

Which ONE of the following describes the electrical power supplies to the following pumps associate with Emergency Diesel Generator, EG-Y-1B:
(1) the Fuel Oil Transfer Pumps, DF-P-1C/D, AND
(2) the Auxiliary Fuel Oil Pump, EG-P-10B

A. (1) The two fuel oil transfer pumps are both AC powered.
(2) The Auxiliary Fuel Oil Pump is AC powered.

B. (1) The two fuel oil transfer pumps are both AC powered.
(2) The Auxiliary Fuel Oil Pump is DC powered.

C. (1) One fuel oil transfer pumps is AC powered, One fuel oil transfer pumps is DC powered.
(2) The Auxiliary Fuel Oil Pump is AC powered.

D. (1) One fuel oil transfer pumps is AC powered, One fuel oil transfer pumps is DC powered.
(2) The Auxiliary Fuel Oil Pump is DC powered.

**32** **ID: 1718191** **Points: 1.00**

Plant conditions:

- Loss of Station Power
- EG-Y-1A, Emergency Diesel Generator, had to be started from the Control Room using the Manual Start Pushbutton.
- EG-Y-1A 'Ready to Load' light is DE-ENERGIZED.
- EG-Y-1A voltage is 4050 V.
- EG-Y-1A frequency is 60.5 Hz.

All control room controls associated with EG-Y-1A were in the ES STANDBY line-up when the diesel was started.

From the list below, identify the ONE (1) action that would ENERGIZE the Ready to Load light under these conditions.

A. Adjust the governor to lower frequency to 60.0 Hz.

B. Energize the synchroscope for EG-Y-1A generator output breaker, G1-02.

C. Adjust the local unit voltage rheostat to obtain an output voltage of 4100 V.

D. Adjust the manual voltage controller on Console Right to obtain an output voltage of 4100 V.

**33** **ID: 1718267** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- OP-TM-AOP-005, RIVER WATER SYSTEM FAILURES, is entered due to reduced river water level and rising river water temperature.
- The crew commenced a plant shutdown.

EVENT:

- Auxiliary Operator report:
  - ISPH Pump Bay Water Level is at 270 foot elevation.
  - River Water temperature is 92 degrees F.

Based on these conditions identify the ONE selection that describes required action(s).

A. Continue the shutdown to HOT SHUTDOWN.

B. Continue the shutdown to COLD SHUTDOWN.

C. Trip the reactor and initiate OP-TM-EOP-001, REACTOR TRIP, only.

D. Trip the reactor, initiate OP-TM-EOP-001, REACTOR TRIP, then trip all four Reactor Coolant Pumps.

**34** **ID: 1718269** **Points: 1.00**

Plant Conditions:
- 100% power with ICS in full auto.
- Main Instrument Air Compressor, IA-P-4, has been secured for repairs.
- Instrument Air Compressors, IA-P-1A and IA-P-1B, are running as required.

EVENTS:

- LOCA
- Loss of Off-Site Power
- 1600 PSI ES Actuation

10 minutes later:

- Due to low instrument air pressure, the CRO starts IA-P-1A.

Cooling for IA-P-1A/B will be aligned to __(1)__ due to the __(2)__.

A. (1) Fire Service Water, (2) ES Actuation Signal.

B. (1) Fire Service Water, (2) Loss of Offsite Power.

C. (1) Secondary Closed Cooling, (2) ES Actuation Signal.

D. (1) Secondary Closed Cooling, (2) Loss of Offsite Power.

**35** **ID: 1718430** **Points: 1.00**

Given the following plant conditions:

- Reactor is at 52% power.
- ICS SG/RX Hand/Auto Station is in HAND.
- All other ICS stations are in AUTOMATIC.

EVENT:
- Group 7 Rod 4 in Quadrant YZ drops fully into the core.

Which of the following indicates the effect on (1) Quadrant Power Tilt in Quadrant YZ and (2) $T_{ave}$ parameter response?

**ASSUME NO OPERATOR ACTIONS.**

A. (1) Negative Quadrant Power Tilt.
(2) $T_{ave}$ lowers and remains low.

B. (1) Negative Quadrant Power Tilt.
(2) $T_{ave}$ lowers and returns to setpoint.

C. (1) Positive Quadrant Power Tilt.
(2) $T_{ave}$ lowers and remains low.

D. (1) Positive Quadrant Power Tilt.
(2) $T_{ave}$ lowers and returns to setpoint.

# EXAMINATION

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**36** **ID: 1737551** **Points: 1.00**

Plant Conditions:

- 60% power with ICS in auto.

EVENT:

- Reactor Building pressure starts to rise.
- RCS pressure starts to lower.
- The reactor is tripped and the IMA's of OP-TM-EOP-001, REACTOR TRIP are performed.

POST TRIP CONDITIONS:

- Reactor Building pressure rises to 4.2 psig.
- RCS pressure lowers to 1675 psig.

Post Trip, RM-A-2, Reactor Building Atmosphere Monitor, __(1)__ be used to determine whether the Reactor Building pressure increase is from the reactor coolant leak because __(2)__.

A. (1) can
(2) there is NOT a 1600 psig ES Actuation Signal

B. (1) can
(2) RM-A-2 is environmentally qualified for accident conditions

C. (1) can NOT
(2) it is isolated from Containment

D. (1) can NOT
(2) RM-A-2 is NOT environmentally qualified for accident conditions

**37** **ID: 1718596** **Points: 1.00**

Sequence of Events:

- Emergency Feedwater Pump, EF-P-2A, is OOS.
- Loss of Offsite Power (LOOP) occurred 10 minutes ago.
- A small break LOCA occurred following the loss of offsite power.
- Emergency Feedwater Pump, EF-P-2B, tripped when it started automatically.
- OP-TM-EOP-001, Reactor Trip, IMAs were performed.
- OP-TM-EOP-006, LOCA Cooldown and OP-TM-AOP-020, Loss of Station Power, actions are in progress.
- 4 psig ESAS was manually actuated.
- RCS temperature is 518°F and SCM has been maintained >25°F throughout the transient.
- Incore Thermocouples are slowly rising.
- Natural Circulation cannot be verified.

Which ONE of the following is the **NEXT** required action?
Reduce OTSG pressure ________________________.

A. while maintaining >750 psig to avoid feedwater isolation.

B. in both OTSGs as low as possible (Atmospheric pressure or vacuum).

C. in both OTSGs to approximately 150 psig at the maximum cooldown rate allowed.

D. so that secondary Tsat is 40 to 60°F lower than incore thermocouple temperature, while maintaining >150 psig.

**38** **ID: 1718720** **Points: 1.00**

Plant conditions:

- 100% power with ICS in full auto.

EVENT:

- Overhead alarm MAP C-1-1, "RADIATION LEVEL HI", just actuated.
- The source of the alarm is RM-A-5, CONDENSER VACUUM PUMP EXHAUST.

The MAP-5 Sampler starts to sample the:

A. steam lines for only iodine when RM-A-5 reaches the ALERT setpoint.

B. steam lines for iodine and tritium when RM-A-5 reaches the ALERT setpoint.

C. condenser offgas for only iodine when RM-A-5 reaches the HI-ALARM setpoint.

D. condenser offgas for iodine and tritium when RM-A-5 reaches the HI-ALARM setpoint.

**39** **ID: 1740782** **Points: 1.00**

Given the following valves:

- IA-P-1A/B: INSTRUMENT AIR COMPRESSOR "A"/"B".
- IA-V-1: SERVICE AIR TO IA X-CONNECT.
- IA-V-26: SECONDARY PLANT IA SUPPLY VALVE.
- IA-V-2104A/B: IA-P-1A/B SUPPLY TO IA SYS BLOCK VALVE.
- IA-V-2133: IA-Q-2 (DRYER) & IA-F-10A/B (POST FILTER) BYPASS VALVE.

Which of the following is the correct sequence of events that automatically occur in the Instrument and Service Systems as air pressure lowers?

| | | |
|---|---|---|
| A. | 85 psig | IA-V-2104A/B OPEN and IA-P-1A/B START. |
| | 80 psig | IA-V-1 OPENS. |
| | 75 psig | IA-V-2133 OPENS. |
| | 60 psig | IA-V-26 CLOSES. |
| B. | 90 psig | IA-V-2133 OPENS. |
| | 85 psig | IA-V-1 OPENS. |
| | 75 psig | IA-V-2104A/B OPEN and IA-P-1A/B START. |
| | 60 psig | IA-V-26 CLOSES. |
| C. | 95 psig | IA-V-2133 OPENS. |
| | 90 psig | IA-V-2104A/B OPEN and IA-P-1A/B START. |
| | 85 psig | IA-V-26 CLOSES. |
| | 70 psig | IA-V-1 OPENS. |
| D. | 100 psig | IA-V-2104A/B OPEN and IA-P-1A/B START. |
| | 90 psig | IA-V-1 OPENS. |
| | 85 psig | IA-V-2133 OPENS. |
| | 70 psig | IA-V-26 CLOSES. |

**40** **ID: 1718755** **Points: 1.00**

REFERENCE PROVIDED

Plant conditions:

- Plant is shutting down to HOT SHUTDOWN.
- Tave is 575 deg-F.
- Pressurizer level is being maintained in accordance with 1102-10, PLANT SHUTDOWN.

Based on the above conditions, which indicated level below would be the **minimum** acceptable pressurizer level?

A. 100".

B. 185".

C. 210".

D. 220".

**41** **ID: 1737589** **Points: 1.00**

Plant Conditions:

- The plant was at 100% power when a control rod in Group 7 dropped.
- During the runback two additional rods in Group 7 become stuck at 88% withdrawn.
- The ICS runback was completed in automatic.

With the above plant conditions the operating crew must _____________within one hour.

A. trip the reactor

B. be in HOT SHUTDOWN

C. verify Axial Power Imbalance is within limits

D. verify SDM $\geq$ 1% delta k/k or initiate boration

**42** **ID: 1718778** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- The Unit has experienced several fuel pin failures.
- A leak must be repaired in the Aux. Bldg.
- The general area dose rate in the location of the repair is 600 mrem/hr.
- In order to reach the location of the repair the worker must transit through a 6 Rem/hr high radiation area for 2 minutes and return via the same path.
- The worker currently has an accumulated annual dose of 400 mrem.

The maximum allowable time that the worker can participate in the repairs and **NOT** exceed the TEDE Administrative Dose Control Limit is _________ minutes (Worker does **NOT** have a High Lifetime Exposure).

A. 100

B. 120

C. 140

D. 160

**43** **ID: 1718893** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Sequence of Events:

T = 1100:
A Reactor Building Purge has commenced.

T = 1200:
The Reactor Building purge is secured due to a minor equipment problem.

T = 1730:
The equipment problem has been fixed and the Reactor Building purge is ready to recommence.

Given the above information and IAW CY-TM-170-2012, RELEASING RADIOACTIVE GASEOUS EFFLUENTS:

The RB Purge may recommence________________.

A. with the same release permit using original air sample data.

B. with the same release permit using updated air sample data.

C. when a new release permit is generated using a new release number with updated air sample data.

D. when a new release permit is generated using the same release number with updated air sample data.

**44** **ID: 1719768** **Points: 1.00**

In accordance with OS-24, which of the following is correct regarding the interruption of the Reactor Trip Immediate Manual Actions?

A. If a Loss of Subcooling Margin is identified, Rule 1 must be performed to trip the reactor coolant pumps, initiate HPI and EFW.

B. If an Excessive Heat Transfer is identified, Rule 3 must be performed to isolate the affected OTSG.

C. If a Lack of Heat Transfer is identified, Rule 4 must be performed to establish feedwater control.

D. If multiple dropped rods are identified, Rule 5 must be completed to initiate Emergency Boration.

**45** **ID: 1719786** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- The reactor has tripped.
- Damage has occurred to the Control Tower, preventing plant control from either the Control Room or the Remote Shutdown Panels.
- On-shift personnel are not responding to any communications.

Based on the above conditions, the priority action to be taken by an Auxiliary Operator is to report to the:

A. EFW area and perform OP-TM-AOP-009, Loss of Plant Control Facilities in order to establish EFW flow.

B. Auxiliary Boilers and perform OP-TM-414-401 (402), to start AS-B-1A (1B) in order to establish Auxiliary Steam for necessary loads.

C. Auxiliary Building and perform Attachment 5 of OP-TM-EOP-020, Cooldown From Outside of Control Room to prevent spurious operation of MOV's.

D. Turbine Building and open AS-V-8 IAW Attachment E of OS-24, Conduct of Operations During Abnormal and Emergency Events, to make Auxiliary Steam available to Gland Steam when Auxiliary Steam pressure is available.

**46** **ID: 1720267** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- A feedwater transient occurs.
- An automatic reactor trip on HIGH RCS temperature fails to occur.
- Main Feedwater remains available.
- The reactor trip and DSS Pushbuttons fail to trip the reactor.
- A CRO opens both the 1L-02 and 1G-02 breakers.
- Several groups of control rods failed to insert.
- Reactor power is 10% and steady.

Based on the above conditions, the CRO must __(1)__ in order to __(2)__ .

A. (1) trip the Main Turbine,
(2) minimize peak RCS pressure

B. (1) trip the Main Turbine,
(2) prevent an overcooling event

C. (1) wait until RCS Pressure is less than 2500 psig, then initiate Emergency Injection HPI/LPI,
(2) inject borated water to the RCS

D. (1) wait until RCS Pressure is less than 2500 psig, then initiate Emergency Injection HPI/LPI,
(2) prevent RCS pressure from exceeding 3000 psig

**47** **ID: 1720289** **Points: 1.00**

A reactor trip from 100% power has occurred.

The following plant conditions exist:

- All four reactor coolant pumps are tripped.
- Emergency Feedwater CANNOT be established to either OTSG.
- Main Feedwater CANNOT be recovered.
- Incore thermocouples are indicating 592°F and rising slowly.
- OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, has been entered.
- SCM is 28°F and lowering.

The CRS must __(1)__ based on __(2)__.

A. (1) go to OP-TM-EOP-009, HPI COOLING, and perform Rule 1 when Subcooling Margin is < 25°F,
(2) SCM approaching 25°F

B. (1) go to OP-TM-EOP-009, HPI COOLING, and perform Rule 1 when Subcooling Margin is < 25°F,
(2) RCS pressure approaching 2450 psig

C. (1) continue with OP-TM-EOP-004 to open the PORV then close it when RCS pressure is 1750 psig,
(2) RCS pressure approaching 2450 psig

D. (1) continue with OP-TM-EOP-004 to open the PORV then close it when SCM approaches 30°F,
(2) SCM approaching 25°F

**48** **ID: 1720350** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Loss of #8 Bus

Based on the Loss of the #8 Bus, the CRS must enter __(1)__ and the CRO is required to make a plant announcement using the __(2)__.

A. (1) OP-TM-AOP-013, LOSS OF 1D 4160V BUS, (2) "RED" plant page and radio

B. (1) OP-TM-AOP-013, LOSS OF 1D 4160V BUS, (2) "GREY" plant page and radio

C. (1) OP-TM-AOP-014, LOSS OF 1E 4160V BUS, (2) "RED" plant page and radio

D. (1) OP-TM-AOP-014, LOSS OF 1E 4160V BUS, (2) "GREY" plant page and radio

**49** **ID: 1720342** **Points: 1.00**

Plant Conditions:

- 40% power with ICS in auto.

EVENT:

- Reactor power is LOWERING.
- RCS pressure is RISING.
- Main Steam Safety Valves are OPEN.
- Turbine Bypass Valves, MS-V-3A-F, and Atmospheric Dump Valves, MS-V-4A/B, are OPEN.
- Indicating lights on Panel SS-1 are GREEN for the breakers for the Middletown 1092, Jackson 1051 and 500 kV tie lines.
- Indicating lights on Panel SS-1 for the Middletown 1091 breaker switches are GREEN and YELLOW.
- Indicating lights for the Main Generator Breakers are RED.
- Generator load is 49 megawatts.

Which of the following procedures must be entered to mitigate this event?

A. MAP-H0101, ICS RUNBACK

B. 1107-11, TMI GRID OPERATIONS

C. OP-TM-AOP-022, LOAD REJECTION

D. OP-TM-AOP-070, PRIMARY TO SECONDARY HEAT TRANSFER UPSET

**50** **ID: 1720492** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- RC3-PR Hot Leg A Narrow Range Channel 1 fails LOW.
- MAP alarm H-3-2, SASS Mismatch is illuminated.
- SASS selector switch for RC3A-PT1 Hot Leg A Narrow Range Channel 1 has the RED and WHITE lights illuminated.
- SASS selector switch for RC3B-PT1 Hot Leg B Narrow Range Channel 1 has only the RED light illuminated.

What is the system response of this failure?

A. All Pressurizer Heater Banks energize.

B. SASS transfers to an alternate channel.

C. One channel of ES actuates on Train 'A' AND Train 'B'.

D. Only SCR controlled banks 1, 2, and 3 Pressurizer Heaters energize.

**51** **ID: 1720498** **Points: 1.00**

Identify the conditions that promote Pool Boiler Condenser Cooling Mode (BCM) of Heat Transfer.

The OTSG thermal center remains __(1)__ the RCP spillover elevation.
RCS liquid level is maintained __(2)__ the secondary level.

A. (1) above
(2) above

B. (1) above
(2) below

C. (1) below
(2) above

D. (1) below
(2) below

**52** **ID: 1720537** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- RPS Channel A in MANUAL BYPASS.

EVENT:

- Vital bus "A" (VBA) losses power.

Given the information above, identify the selection below that describes (1) the number of remaining channels required to trip and (2) the degree of redundancy.

A. (1) ONE (2) ONE

B. (1) ONE (2) TWO

C. (1) TWO (2) ONE

D. (1) TWO (2) TWO

**53** **ID: 1720550** **Points: 1.00**

Initial conditions:

- 100% Reactor power.
- AH-E-9A, PENETRATION COOLING FAN, is in Normal-after-start.
- AH-E-9B, PENETRATION COOLING FAN, is in Normal-after-stop.

Event:

- Reactor Trip occurs.
- 4# ES actuation occurs (both trains).
- HVA-2-1, PENETRATION COOLING AIR TEMP HI, alarms.
- Recorder TR-805, PENETR COOL AIR TEMPERATURE RECORDER, points 1-25, all read >205°F and rising.

What **MINIMUM** action(s) must be taken to lower penetration temperatures?

A. Start AH-E-9B ONLY.

B. Restart AH-E-9A ONLY.

C. Bypass both 4# ES signals and then start AH-E-9B.

D. Ensure AH-D-89, AH-E-9A/B OUTSIDE AIR INLET, and AH-D-90, AH-E-9A/B TURBINE BLDG INLET, are open.

**54** **ID: 1720590** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Due to a fault, the feeder breaker to 1D 4160V bus trips open.
- Emergency Diesel Generator, EG-Y-1A is powering the 1D 4160V bus.

The 1N 480V Bus is __(1)__ because __(2)__.

A. (1) energized
(2) the 1N 480V bus automatically repowers from the 1L 480V bus crosstie

B. (1) energized
(2) the 1N 480V bus is repowers when EG-Y-1A repowers the 1D 4160V bus

C. (1) de-energized
(2) the UV results in a 1N 480V bus trip and lockout

D. (1) de-energized
(2) the UV results in a 1N 480V bus trip with NO lockout

**55** **ID: 1720616** **Points: 1.00**


Identify the correct reading for RM-G-23, RB High Range Radiation Monitor.

A. $1X10^3$ R/h

B. $1X10^3$ mr/h

C. $2X10^2$ R/h

D. $2X10^2$ mr/h

**56** **ID: 1740713** **Points: 1.00**

Plant conditions:

- Reactor at 100% power with ICS in full auto.
- Turbine Bypass Valves for the 'A' OTSG, MS-V-3D, MS-V-3E and MS-V-3F are closed in HAND for ICS module replacement.

EVENT:

- Reactor Trip

The 'A' OTSG will be controlled at __(1)__.

A. 895 psig

B. 965 psig

C. 1010 psig

D. 1026 psig

**57** **ID: 1720668** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Sequence of Events:

- The 'B' OTSG develops a steam leak in the Reactor Building.
- The crew manually trips the reactor based on rising Reactor Building pressure.
- A 4 PSIG ESAS actuation occurs.
- Current plant conditions:
  - RCS $T_{ave}$ is 560°F and lowering slowly.
  - OTSG 1A level is 86 inches on the Startup Range and slowly lowering.
  - OTSG 1B level is 86 inches on the Startup Range and slowly lowering.
  - OTSG 1A pressure is 985 psig and steady.
  - OTSG 1B pressure is 950 psig and lowering very slowly.
  - Main Feedwater flow to OTSG 1A is $0.3 \times 10^5$ lbm/hr.
  - Main Feedwater flow to OTSG 1B is $1.9 \times 10^5$ lbm/hr

The 'B' OTSG must be isolated in accordance with which procedure?

A. OP-TM-EOP-001, REACTOR TRIP

B. OP-TM-EOP-010, Guide 12, RCS STABILIZATION

C. OP-TM-AOP-051, SECONDARY SIDE HIGH ENERGY LEAK

D. OP-TM-EOP-003, EXCESSIVE PRIMARY TO SECONDARY HEAT TRANSFER

**58** **ID: 1720680** **Points: 1.00**

Plant Conditions:
- 65% reactor power.
- ULD is in manual, all other ICS stations are in auto.
- Both SASS modules are de-energized.

EVENT:

- RC-P-1C breaker opens.

The output of RC-12-TAS Tave Auto/Manual Switch will __(1)__ because __(2)__.

A. (1) remain at the Loop A&B Average
(2) SASS is inoperable

B. (1) remain at the Loop A&B Average
(2) reactor power is below the ICS Runback setpoint

C. (1) swap to Loop A Average
(2) the RC-P-1C breaker is open

D. (1) swap to Loop A Average
(2) this loop has the highest RCS flow

**59** **ID: 1720699** **Points: 1.00**

With the Enable/Defeat switch for DC-V-2A/2B Decay Closed Cooler Inlet and DC-V-65A/65B Decay Closed Cooler Bypass placed in the DEFEAT position, this would result in DC-V-2A/2B __(1)__ AND DC-V-65A/65B __(2)__ .

A. (1) full closed
(2) full open

B. (1) full closed
(2) full closed

C. (1) full open
(2) full closed

D. (1) full open
(2) full open

**60** **ID: 1720769** **Points: 1.00**

Which one of the following Alternate Emergency Boration source tanks is required in accordance with 1301-1, SHIFT AND DAILY CHECKS.

Boric Acid Mix Tank (BAMT)
Borated Water Storage Tank (BWST)
Reactor Coolant Bleed Tanks (RCBTs)
Reclaimed Boric Acid Tanks (RBATs)

A. BWST OR RCBT

B. BAMT OR RCBT

C. BAMT OR RBAT "A" OR RBAT "B"

D. BWST OR RBAT "A" OR RBAT "B"

# EXAMINATION

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**61** **ID: 1720771** **Points: 1.00**

During normal 100% power operations, the steam supply to the Main Feedwater Pumps is the outlet of the __(1)__ , and shortly following a Reactor/Turbine trip, the source of steam to the Main Feedwater Pumps is __(2)__.

A. (1) "D" AND "F" Moisture Separators
(2) Main Steam

B. (1) "D" AND "F" Moisture Separators
(2) Auxiliary Steam

C. (1) HP Turbine prior to "C" AND "D" Moisture Separators
(2) Main Steam

D. (1) HP Turbine prior to "C" AND "D" Moisture Separators
(2) Auxiliary Steam

**62** **ID: 1720783** **Points: 1.00**

Plant Condition:

- 100% power with ICS in full auto.

EVENT

- Loss of Offsite Power

Subsequently:

- All 8 (eight) HSPS switches placed in DEFEAT.
- No other operator actions have been taken.

What is the setpoint from HSPS for OTSG level control?

A. 0" on the startup range.

B. 25" on the startup range.

C. 50% on the operating range.

D. 75-85% on the operating range.

**63** **ID: 1724247** **Points: 1.00**

Plant Conditions:

- 40% power with ICS in auto.
- Main Condenser Tube Leak on the "A" side.
- Circulating Water Pumps, CW-P-1A, CW-P-1B, CW-P-1C are shutdown.
- Loop Cross Connect Valves, CW-V-4A/4B are closed.

EVENT:

- CW-P-1D Trips.
- Main Condenser vacuum starts to lower.
- CRS directs the CRO to trip the Main Turbine.
- Main Condenser vacuum lowers to 25" Hg and stabilizes.

Based on the conditions, the reactor has __(1)__ and OTSG pressure control is being controlled by the __(2)__.

A. (1) tripped
(2) Turbine Bypass Valves

B. (1) tripped
(2) Atmospheric Dump Valves

C. (1) stabilized at ~18% power
(2) Turbine Bypass Valves

D. (1) stabilized at ~18% power
(2) Atmospheric Dump Valves

**64** **ID: 1736357** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Electrical load reducing, with a constant reactor power.
- OTSG 1A steam pressure is 870 psig and decreasing.
- RB pressure is 0.3 psig and steady.
- Total FW flow is 10.2 E6 lbm/hr.
- PLF-1-9 and PLF-1-10 for Intermediate Bldg Fire AND Intermediate Bldg Trouble
- AH-E-73 (Intermediate Bldg.) discharge high temperature alarm.
- PPC Point A0331 TURB DRIV EF-P1 BRG COOL WTR OUT is in HI-2 Alarm
- RM-A-2G reading normal.

What action is required for the above event?

A. Dispatch personnel to look for the steam leak.

B. Manually trip the Reactor AND go to OP-TM-EOP-001.

C. Reduce power to < 45% and manually trip the Main Turbine.

D. Initiate a Plant Shutdown IAW 1102-4, POWER OPERATION.

**65** **ID: 1718268** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- Building Spray Pump, BS-P-1B, is out of service.

SEQUENCE OF EVENTS:

1. LOCA
2. RCS pressure lowers to 890 psig.
3. Reactor Building Pressure rises to 32 psig.
4. 1S 480V ES Bus trips.

Post Accident Reactor Building cooling is __(1)__, because __(2)__.

A. (1) adequate
(2) one Building Spray Pump and one Reactor Building Emergency Cooling Fan (AH-E-1) are operating

B. (1) adequate
(2) one Building Spray Pump and two Reactor Building Emergency Cooling Fans (AH-E-1) are operating

C. (1) NOT adequate
(2) only one Building Spray Pump and one Reactor Building Emergency Cooling Fan (AH-E-1) are operating

D. (1) NOT adequate
(2) only one Building Spray Pump and two Reactor Building Emergence Cooling Fans (AH-E-1) are operating

**66** **ID: 1736429** **Points: 1.00**

Plant Conditions:

- 90% power with ICS in auto.

EVENT:

- PLB-1-7 INSTRUMENT AIR PRESS LOW TURBINE AREA comes in.
- PI-1403 Secondary IA pressure reads 59 psig.
- PI-222 Primary Instrument Air pressure reads 90 psig.

Based on the above plant conditions, the initial action the operating crew must perform is:

A. Trip the Reactor.

B. Initiate a plant shutdown.

C. Dispatch a NLO to isolate Primary IA from Secondary IA.

D. Start Instrument Air Compressors, IA-P-1A and IA-P-1B from the Control Room.

**67** **ID: 1736434** **Points: 1.00**

Plant Conditions:

- The Plant has been shutdown for 24 days.
- Reactor Vessel level is being lowered in support of internal Reactor Vessel work.
- The "B" Train of Decay Heat Removal (DHR) is operating with a flow rate of 1,400 gpm.

EVENT:

- PLB-5-5 1B DECAY HEAT REMOVAL COMPARTMENT LEAK DETECT comes in.
- Decay Heat Pump, DH-P-1B, suction line temperature indicator has risen 10.5°F.
- RCS Cold Leg level is 15 inches above centerline and slowly lowering.
- DH-P-1B motor amperage and flow is significantly oscillating.

Based on the above conditions, which ONE of the following actions must be taken FIRST?

A. Start "A" Train of Decay Heat Removal.

B. Lower flowrate from DH-P-1B to less than 1,200 gpm.

C. Immediately place DH-P-1B in the "Pull-to-Lock" position.

D. Raise RCS level until Decay Heat removal Pump operating indications are normal.

**68** **ID: 1736443** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- MU-P-1A (Make-up Pump) is supplying normal make-up and Seal Injection.
- MU-P-1B is in Normal-After Stop.

SEQUENCE OF EVENTS:

- Loss of Coolant Accident.
- RCS pressure lowers to 1500 psig.
- RB pressure is 5 psig.
- Loss of Offsite Power.
- MU-P-1A Trips.

Based on the above conditions, if MU-P-1B is ES Selected using the 43 Selector Switch on the 1E 4160V Bus, MU-P-1B would __(1)__ and MU-P-1C would __(2)__.

A. (1) start and immediately trip, (2) trip.

B. (1) start and immediately trip, (2) continue to run.

C. (1) start and continue to run, (2) continue to run.

D. (1) start and continue to run, (2) trip.

**69** **ID: 1736454** **Points: 1.00**

Plant Conditions:

- 50% power with ICS in auto.

EVENT:

- MAP G-2-1 CRD PATTERN ASYMMETRIC alarms.
- Red Fault Indication lit on the Position Indication Panel (PIP).
- The Asymmetric Control Rod Fault indication is currently lit on the Diamond Control Panel.

These indications would be caused by a deviation of an individual rod position of ___(1)___from the group average, which ___(2)___, and would require entry into OP-TM-AOP-062, INOPERABLE ROD.

A. (1) 7 inches
(2) excludes the faulted rod

B. (1) 7 inches
(2) includes the faulted rod

C. (1) 9 inches
(2) excludes the faulted rod

D. (1) 9 inches
(2) includes the faulted rod

**70** **ID: 1736470** **Points: 1.00**

What is minimum personnel that must be in the Control Room in accordance with OP-TM-112-101-1002, SHIFT STAFFING REQUIREMENTS, when the RCS temperature is greater than 200F?

A. One SRO or one RO

B. One SRO AND one RO

C. One SRO AND two ROs

D. One SRO OR one RO AND one STA

**71** **ID: 1736534** **Points: 1.00**

In accordance with Tech Specs, the Minimum Conditions for Criticality (Excluding Low Power Physics Testing) requires the RCS Temperature to be __(1)__ and the Safety Rod Groups to be __(2)__ prior to any other reduction in shutdown margin.

A. (1) ≥ 532F (2) fully withdrawn

B. (1) ≥ 532F (2) fully inserted

C. (1) >525F (2) fully withdrawn

D. (1) >525F (2) fully inserted

**72** **ID: 1737637** **Points: 1.00**

A surveillance for an ES pump is in progress.
The analog suction pressure gauge, normally isolated except for this surveillance, is broken.
The suction pressure is used to calculate pump head, which is a required parameter for the surveillance test.
An alternate instrument is available, but it is not listed in the surveillance procedure.

In accordance with ER-TM-321-1041, TMI-1 IST PROGRAM REQUIREMENTS, which one of the following identifies the required actions with respect to this surveillance?

A. Continue the surveillance. Annotate in the procedure that the alternate instrument is used.

B. Stop the surveillance. Evaluate if the alternate gauge can be used.

C. Stop the surveillance. The surveillance may only be run with the installed gauge.

D. Stop the surveillance. Perform an IC to allow use of the alternate instrument, then continue the surveillance.

**73** **ID: 1736792** **Points: 1.00**

Plant Conditions:

- OP-TM-EOP-005, OTSG TUBE LEAKAGE has been entered for a 'A' OTSG tube rupture.
- Reactor is tripped.

'A' OTSG must be isolated when ___(1)___ AND ___(2)___.

A. (1) RCS pressure is less than 1000 psig,
(2) BWST level < 22 feet

B. (1) OTSG pressure is less than 1000 psig,
(2) BWST level < 22 feet

C. (1) RCS pressure is less than 1000 psig,
(2) Offsite integrated dose approaches 500 mRem CDE (Thyroid)

D. (1) OTSG Pressure is less than 1000 psig,
(2) Offsite integrated dose approaches 500 mRem CDE (Thyroid)

**74** **ID: 1740715** **Points: 1.00**

Plant Conditions:

- Rx vessel head being de-tensioned.
- "A" Decay Heat Removal (DHR) train in service.
- "B" DHR train in standby.

Event:

- "A" Decay Closed Cooling Tank, DC-T-1A, rising at rate equivalent to 5 gpm.
- "A" Decay Closed System Inline Rad Monitor, RM-L-2, counts have risen.

Based on the conditions above which ONE of the below actions must be taken?

A. Isolate DC-T-1A vent.

B. Initiate Containment Isolation.

C. Swap cooling to "B" DHR, and isolate the "A" DHR.

D. Transfer heat removal to the OTSGs and secure DHR.

**75** **ID: 1736802** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Which on of the following identifies the status of the Reactor Building Hydrogen Monitor Recorders?

They are both __(1)__ and they are located on Control Room __(2)__.

A. (1) ON (2) the H&V Panel.

B. (1) ON (2) Panel Left.

C. (1) OFF (2) the H&V Panel.

D. (1) OFF (2) Panel Left.

**76** **ID: 1737249** **Points: 1.00**

Plant Conditions:

- Reactor has tripped on RCS low pressure.
- Pressurizer level is 370 inches and slowly increasing.
- RCS pressure is 1500 psig and slowly lowering.
- Core exit temperature is 556 degrees F and stable.
- Reactor Building sump level is 38 inches and increasing.

(1) What is the location of the RCS Leak?
(2) Which of the following combinations of procedures provide the correct guidance to cooldown the plant?

A. (1) Leak in the RCS Cold Leg
(2) Enter OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN, and use OP-TM-EOP-006, LOCA COOLDOWN to cooldown the plant.

B. (1) Leak in the RCS Cold Leg
(2) Enter OP-TM-AOP-050, REACTOR COOLANT LEAKAGE and use 1102-11, PLANT COOLDOWN to cooldown the plant.

C. (1) Leak in the Pressurizer Steam Space
(2) Enter OP-TM-AOP-043, LOSS OF PRESSURIZER, and use 1102-11, PLANT COOLDOWN to cooldown the plant.

D. (1) Leak in the Pressurizer Steam Space
(2) Enter OP-TM-AOP-050, REACTOR COOLANT LEAKAGE and use 1102-11, PLANT COOLDOWN to cooldown the plant.

**77** **ID: 1700250** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- MAP C-3-2, IC SURGE TANK LEVEL HI/LO, alarm in.
- RM-L-9, Intermediate Closed Cooling Water Radiation Monitor, counts are rising.
- IC-T-1, Intermediate Closed Cooling Water Surge Tank, level is off-scale high, and an attempt to bring IC-T-1 back on scale has failed.
- OP-TM-AOP-050, REACTOR COOLANT LEAKAGE is entered.

Current plant parameters:

- Pressurizer level is 220 inches and steady.
- MU Tank level is 86 inches and slowly lowering.
- Reactor Building pressure is 0.2 psig and steady.

Which ONE of the following choices identifies the correct procedure and actions that must be implemented?

A. 1) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
2) Initiate a plant shutdown to be in HOT SHUTDOWN within 24 hours

B. 1) OP-TM-MAP-C0302, IC SURGE TANK LEVEL HI/LO
2) Trip the reactor and close IC-V-2 and IC-V-3

C. 1) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
2) Trip the reactor and close IC-V-2 and IC-V-3

D. 1) OP-TM-MAP-C0302, IC SURGE TANK LEVEL HI/LO
2) Initiate a plant shutdown to be in HOT SHUTDOWN within 24 hours

**78** **ID: 1685279** **Points: 1.00**

Plant conditions:

- Reactor is at 100% power
- DC-P-1A is tagged out for a bearing replacement.
- MU-P-1A and MU-P-1C are ES selected and operable.
- The Secondary NLO has just reported the fuel rack for the SBO Diesel was accidentally broken in the **tripped position** by a contract worker climbing over the lever.
- Maintenance says the repair will take 4 hours.

EVENT:

- One-half of the overhead lights go out, and do NOT come back on.
- You observe the following MAPs actuate:
  - B-1-1, 4KV ES FDR BKR TRIP.
  - B-2-1, 4KV ES BUS UV/OV.
  - B-1-2, 4KV ES MOTOR TRIP.
  - F-1-5, RCP SEAL TOT INJECT FLOW HI/LO.

Based on these conditions, which ONE of the following actions must be taken?

A. Initiate Reactor shutdown within 1 hour.

B. Swap MU-P-1A Cooling from DC to NS IAW OP-TM-543-439.

C. Initiate OP-TM-864-901 SBO Diesel Generator (EG-Y-4) Operations.

D. Ensure Emergency Diesel Generator EG-Y-1B restored to operable status within 7 days.

**79** **ID: 1685296** **Points: 1.00**

Plant conditions:

- Reactor power is 35%
- Turbine load is 300 MWe.
- Main Feedwater Pump 1A (FW-P-1A) is tagged out of service.
- All other equipment lineups are normal.

Event:

- MAP Alarm M-1-7, FWP-1B TRIP, actuates.
- Reactor power and turbine load are stable.
- OTSG "A" and "B" levels are lowering.

Given the above information, which ONE of the following actions is required?

A. Initiate a MANUAL reactor trip and enter OP-TM-EOP-001, REACTOR TRIP.

B. Ensure MAP H-1-1, ICS RUNBACK, actuates and implement 1102-4, POWER OPERATIONS.

C. Rapidly reduce main turbine load until reactor power is less than 7% and implement OP-TM-424-901, EMERGENCY FEEDWATER.

D. Trip the main turbine and reduce reactor power to less than 7% and then stabilize the plant in accordance 1102-4, POWER OPERATIONS.

# EXAMINATION

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**80** **ID: 1685310** **Points: 1.00**

Plant Conditions:

- RCS Heat up in progress.
- RCS Temperature is 290°F and rising.
- DC Distribution Panel 1M transfer switch is selected to "A" DC Distribution System and the switch on PCR is in "Auto".
- Makeup Pump 1B is supplying seal injection.

Event:

- The following alarms have actuated simultaneously:
  - AA-3-2 7 KV Bus Trouble.
  - AA-3-3 4 KV BOP Bus Trouble.
  - AA-3-5 480V BOP Bus Trouble.
  - A-1-7 Battery 1A Discharging.
  - A-2-7 Battery Charger 1A/1C/1E Trouble.
  - A-3-7 Inverter 1A/1C/1E Trouble.
  - B-3-1 4KV ES Bus Trouble.
  - PRF1-1-1 CRDM Brkr Test Trouble.
  - H&V A 4-2 Contr Bldg Bat Chgrs A Damper Tbl Fire-Smoke.
  - Loss of numerous breaker status lights at control switches.

Based on these conditions, identify the ONE selection below that describes the:
(1) Applicable procedure to respond to these conditions, and
(2) Impact on the ability to operate safety related equipment.

A. (1) OP-TM-AOP-023, "A" DC System Failure.
(2) Makeup Pump 1B breaker must be racked out due to LTOP concerns.

B. (1) OP-TM-AOP-023, "A" DC System Failure.
(2) Pressurizer Level must be verified $\leq$ 100 inches because the PORV setpoint is greater than 592 psig.

C. (1) OP-TM-AOP-024, "B" DC System Failure.
(2) Makeup Pump 1B breaker must be racked out due to LTOP concerns.

D. (1) OP-TM-AOP-024, "B" DC System Failure.
(2) Pressurizer Level must be verified $\leq$ 100 inches because the PORV setpoint is greater than 592 psig.

**81** **ID: 1720494** **Points: 1.00**

Plant conditions:

- A loss of vacuum of causes both Main Feedwater pumps to trip.
- Due to equipment failures EF-P-2A, Motor Driven Emergency Feedwater Pump, is the only operating pump to supply FW to the OTSGs.
- RCS is at HSD conditions:
  - RCS T-Hot is 532 degrees F and rising.
  - RCS pressure is 2150 psig and rising.
  - OTSG pressures are 885 psig and steady.
  - OTSG levels are 25" and steady.
- FW-P-1A vacuum is being restored.

EVENT:

- EF-P-2A trip.
- Incore temperatures are rising.

Based on these conditions identify the ONE selection below that describes:
(1) Method of core cooling to be established.
(2) Applicable procedure.

A. (1) Rapid cooldown to LPI injection.
(2) OP-TM-EOP-002, LOSS OF 25 DEGREES F SUBCOOLING MARGIN.

B. (1) Condensate booster pump feed.
(2) OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER.

C. (1) Rapid cooldown to LPI injection.
(2) OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER.

D. (1) Condensate booster pump feed.
(2) OP-TM-EOP-002, LOSS OF 25 DEGREES F SUBCOOLING MARGIN.

**82** **ID: 1737141** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- LOCA requiring a plant trip.
- Pressurizer Level cannot be maintained without HPI.
- OP-TM-EOP-006 LOCA COOLDOWN is in progress.

(1) Which Reactor Coolant Pumps must get secured?
(2) What is the OP-TM-EOP-006 basis for securing them?

A. (1) RC-P-1A and RC-P-1B
(2) Minimize the RCS inventory loss due to the LOCA.

B. (1) RC-P-1A and RC-P-1B
(2) The subsequent cooldown could result in "core lift".

C. (1) RC-P-1C and RC-P-1D
(2) Minimize the RCS inventory loss due to the LOCA.

D. (1) RC-P-1C and RC-P-1D
(2) The subsequent cooldown could result in "core lift".

**83** **ID: 1737631** **Points: 1.00**

The specific activity of the primary and secondary coolant are as follows:

| | Two days ago at 1200 | Yesterday at 1200 | Today at 1200 |
|---|---|---|---|
| RCS Dose Equivalent I-131 | 0.50 microcuries/gram | 0.53 microcuries/gram | 0.52 microcuries/gram |
| OTSG Dose Equivalent I-131 | 0.05 microcuries/gram | 0.053 microcuries/gram | 0.052 microcuries/gram |

The ____(1)_____ is out of specification, and the basis of the specification is to maintain dose within the limits from ____(2)_____.

A. (1) RCS
(2) a tube rupture accident

B. (1) OTSG
(2) a steam line break accident

C. (1) RCS
(2) baseline OTSG tube leakage

D. (1) OTSG
(2) normal condenser offgas

**84** **ID: 1699987** **Points: 1.00**

The following plant conditions exist:

- A small break LOCA is in progress.
- The Reactor tripped, and ESAS actuated on low RCS pressure.
- A loss of off-site power (LOOP) occurred.
- Sub cooling margin was lost and is currently –1°F on the PT plot.
- Incore temperatures are rising slowly.
- RCITS shows a large steam void in each RCS hot leg.
- Current cooldown rate is 45°F / hr.

Which ONE of the following describes the correct actions the crew must take to ensure adequate core cooling?

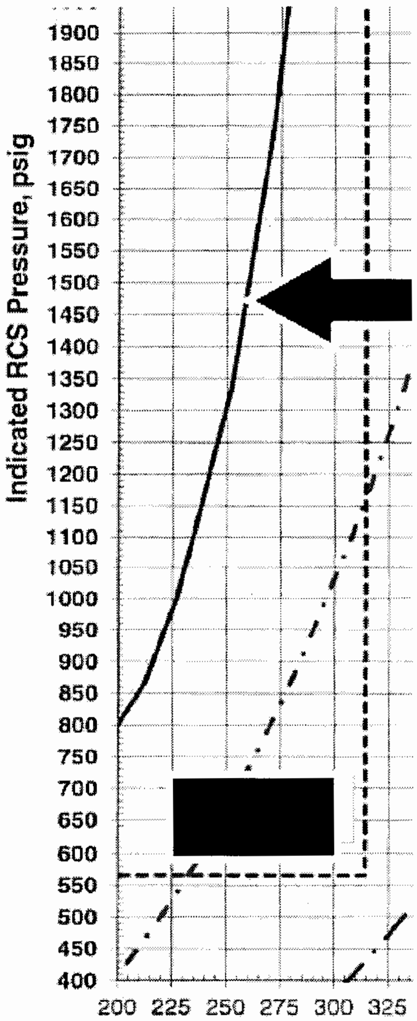
A. OTSG level must be maintained above 50%, with OTSG saturation temperature from 40°F to 60°F below RCS cold leg temperature to promote boiler-condenser cooling. No additional action is required due to the steam void.

B. OTSG level must be maintained above 50%, with OTSG saturation temperature from 40°F to 60°F below RCS cold leg temperature to allow single-phase natural circulation. The steam void must be vented from the hot leg.

C. OTSG level must be maintained 75% - 85%, with OTSG pressure at least 100 psig below RCS pressure to promote boiler-condenser cooling. No additional action is required due to the steam void.

D. OTSG level must be maintained 75% - 85%, with OTSG pressure at least 100 psig below RCS pressure to allow single-phase natural circulation. The steam void must be vented from the hot leg.

**85** **ID: 1720678** **Points: 1.00**

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REFERENCE PROVIDED

Plant conditions:

- Cooldown in progress in accordance with OP-TM-EOP-006, LOCA COOLDOWN.

EVENT:

- Due to a plant upset, RCS temperature is approaching the curve on Figure 1, RCS PRESSURE-TEMPERATURE LIMITS.

The operating crew must ensure open the ___(1)___ to prevent exceeding the___(2)___, in accordance with Guide 23, RCS PRESSURE AND TEMPERATURE LIMITS.

A. (1) PORV
(2) TS NDT Curve

B. (1) PORV
(2) PZR SURGE LIMIT Curve

C. (1) Spray Valve
(2) TS NDT Curve

D. (1) Spray Valve
(2) PZR SURGE LIMIT Curve

**86** **ID: 1740783** **Points: 1.00**

Plant Conditions:

- Refueling Operations in progress.
- DH Loop A in operation.

Event:

- DH-P-1A trips on overcurrent.
- Incore and RCS temperatures have risen from 134°F to 145°F and are currently steady.

Based on these conditions, identify the ONE selection below that describes:

(1) Whether a Reactor Operating Mode change has occurred.
(2) Procedure to be used to respond to the event.

A. (1) A Reactor Operating Mode change has occurred.
(2) EOP-030, Loss of Decay Heat Removal.

B. (1) A Reactor Operating Mode change has NOT occurred.
(2) EOP-030, Loss of Decay Heat Removal.

C. (1) A Reactor Operating Mode change has occurred.
(2) OP-TM-212-901, Emergency DHR Operations.

D. (1) A Reactor Operating Mode change has NOT occurred.
(2) OP-TM-212-901, Emergency DHR Operations.

# EXAMINATION

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**87** **ID: 1720770** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- A worker reported a fire in the Relay Room and evacuated the area.
- The operating crew has entered OP-TM-EOP-020, COOLDOWN OUTSIDE OF CONTROL ROOM.
- The crew has tripped both Main Feedwater Pumps, all running Condensate Booster Pumps, and all running Condensate Pumps.

(1) What attachment will the Primary Safe Shutdown NLO perform?
(2) What event will this attachment prevent or terminate?

A. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Terminate uncontrolled HPI due to a spurious "A" train ES actuation.

B. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Prevent an overcooling event by preventing MS-V-2A/B (Isolations to EF-P-1, TBV's and ADVs) from spuriously opening.

C. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent the Reactor Coolant Pumps from spuriously starting.

D. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent Reactor Coolant pumps from operating with no seal injection.

**88** **ID: 1720789** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Main and Emergency Feedwater are both lost.
- HPI/PORV cooling is in progress.

POST EVENT:

- Both Main and Emergency Feedwater are restored.
- One Reactor Coolant Pump is running in each loop.
- Both OTSG's are intact and dry.
- PPC Points C4015 and C4016 (OTSG tube-to-shell differential temperatures) are at +70F.

In accordance with Guide 13, FEEDING A DRY OR DEPRESSURIZED OTSG, __(1)__ is the preferred feed source because it will __(2)__.

A. (1) Main Feedwater
(2) cool down the OTSG tubes faster

B. (1) Main Feedwater
(2) minimize the tensile loading on the OTSG tubes

C. (1) Emergency Feedwater
(2) cool down the OTSG tubes faster

D. (1) Emergency Feedwater
(2) minimize the tensile loading on the OTSG tubes

**89** **ID: 1737969** **Points: 1.00**

Plant conditions:

- 100% power with ICS in full auto.
- Backup Instrument Air Compressor IA-P-2A is tagged out of service for maintenance.
- Main Vacuum Pump VA-P-1A is operating.

Sequence of events:

| Time | Event |
|---|---|
| 0100 | Local fire (now extinguished) renders Service Air Compressor's SA-P-1A and SA- P-1B inoperable. |
| 0200 | MAP PLB-1-6 IA-P-4 IA-Q-2 TROUBLE actuates. |
| 0210 | Instrument Air Compressor IA-P-1A trips on motor overcurrent. |
| 0220 | Instrument Air Compressor IA-P-1B trips on motor overcurrent. |
| 0230 | The following valid alarms actuate:<br>PLB-1-7 Instrument Air Press Low Turbine Area.<br>PLB-1-8 Station Service Air Press Low.<br>PLB-2-7 Instrument Air Press Low Aux Bldg Area. |
| 0240 | Current Conditions:<br>• All Instrument and Service Air System pressure indications are now at 75 psig, reducing at 1 psi per minute.<br>• Main Condenser vacuum is 27 inches Hg, reducing at 0.2 inches Hg per minute.<br>• Instrument Air Compressor IA-P-4 filter and dryer d/p at 22 psid |

What action is required for these conditions that will prevent entry into Technical Specification LCO 3.4, Decay Heat Removal Capability?

A. Trip the reactor.

B. Take manual control of MU-V-20.

C. Open VA-V-5A vacuum pump suction valve.

D. Ensure open IA-Q-2 bypass valve IA-V-2133 and open IA-V-2124 (Pre-filter Bypass Valve).

**90** **ID: 1737999** **Points: 1.00**

Plant Conditions:

- The plant is shut down in a Refueling Outage.
- Core off-load in progress.
- Containment Integrity is NOT being maintained.
- RB Purge is in progress IAW OP-TM-823-408 RB Purge - RB Doors and/or Equipment Hatch Open.

Event:

- The Fuel Handling SRO reports that a Fuel Assembly has been dropped and damaged in the Fuel Transfer Canal.
- MAP C-1-1 RADIATION LEVEL HI, alarms.
- RB Purge Exhaust Duct Monitor RM-A-9 is off-scale high.
- RB Purge Exhaust Duct Monitor RM-A-9 Hi-Hi is 3000 CPM.
- NI-11 counts have gone up and steadied out at 30 cps.

In accordance with OP-TM-MAP-C0101, all personnel are __(1)__ to be evacuated from the reactor building, and the crew must __(2)__ the RB purge is secured in accordance with OP-TM-244-911, CONTAINMENT CLOSURE.

A. (1) required
(2) ENSURE

B. (1) required
(2) VERIFY

C. (1) NOT required
(2) ENSURE

D. (1) NOT required
(2) VERIFY

**91** **ID: 1700311** **Points: 1.00**

Plant conditions:

- The crew is performing a plant startup per 1102-02, Plant Startup.
- Deboration to the Target Critical Boron Concentration has just been completed.
- The crew is awaiting Plant Manager authorization to startup the reactor.

EVENT:

A check of the Backup Incore Thermocouple Readout (BIRO) reveals that the following Thermocouples have failed:

4-E
5-G
7-B
13-C

Given these current conditions, which ONE of the following identifies whether the LCO for Tech Spec 3.5.5, Accident Monitoring Instrumentation, is currently satisfied, and if not, also identifies the MINIMUM actions required to satisfy the LCO without reliance on any action statement?

A. LCO 3.5.5 is met;
No action statements are required to be entered.

B. LCO 3.5.5 is NOT met; AND
Performing maintenance and declaring Thermocouple 13-C OPERABLE will allow all actions statements to be exited.

C. LCO 3.5.5 is NOT met; AND
Performing maintenance and declaring Thermocouple 4-E OPERABLE will allow all actions statements to be exited.

D. LCO 3.5.5 is NOT met; AND
Performing maintenance and declaring Thermocouple 4-E and 7-B OPERABLE will allow all actions statements to be exited.

**92** **ID: 1736504** **Points: 1.00**

Plant conditions:

- 100% power with ICS in full Auto.
- Condensate Pump CO-P-1C is tagged out for maintenance.

EVENT:

- The Secondary NLO reports that vibrations on Condensate Pump CO-P-1A have increased since his last round and a high pitched noise seems to be coming from the lower motor bearing.
- Maintenance confirms that the lower bearing is failing and recommends CO-P-1A be stopped within the next hour to avoid shaft damage.

The operating crew must reduce power to __(1)__, and secure CO-P-1A in accordance with __(2)__.

A. (1) < 665 MWe
(2) 1102-4, POWER OPERATIONS

B. (1) < 665 MWe
(2) OP-TM-421-430, REMOVING CO-P-1A FROM SERVICE

C. (1) < 50% power
(2) 1102-4, POWER OPERATIONS

D. (1) < 50% power
(2) OP-TM-421-430, REMOVING CO-P-1A FROM SERVICE

**93** **ID: 1736533** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- An approved radioactive liquid release is in progress.

EVENT:

- Annunciator C-1-1 "RADIATION LEVEL HI" is in alarm.
- Radiation Monitor RM-L-6, RAD WASTE DISCHARGE, has failed (pegged high, off-scale).
- WDL-P-14A/B discharge valve to the MDCT, WDL-V-257, is stuck open and cannot be closed.

Identify the one selection below that describes:
(1) The correct action, and
(2) The timeclock associated with RM-L-6.

A. (1) Terminate the release and notify Radiation Protection.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 14 days.

B. (1) Terminate the release and notify Radiation Protection.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 30 days.

C. (1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 14 days.

D. (1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 30 days.

**94** **ID: 1700144** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Per Technical Specifications, which of the following is true regarding makeup tank level and pressure?

Extended operation inside the ___(1)___ is not permitted due to prevention of ___(2)___.

A. (1) Restricted Region
(2) a loss of NPSH when 2 Makeup Pumps are running.

B. (1) Low NPSH region
(2) a loss of NPSH when 2 Makeup Pumps are running.

C. (1) Restricted Region
(2) gas entrainment into the makeup pumps in the event of an emergency injection on a Large Break LOCA.

D. (1) Low NPSH region
(2) gas entrainment into the makeup pumps in the event of an emergency injection on a Large Break LOCA.

**95** **ID: 1700178** **Points: 1.00**

Plant conditions:

- The plant is in refueling outage with fuel handling operations in the RB in progress
- The Main Bridge Operator has lowered an assembly that is NOT an anti-straddle type assembly into the core location with open water on one side only.
- The ZZ tape reading for the assembly is approximately 2" high with full down load cell indication

With the above indications the assembly is resting on the ________________.

A. grid of an adjacent assembly and can be shaken while moving through the grid region as the assembly is lowered

B. grid of an adjacent assembly and the mast can be rotated to reposition the assembly prior to lowering

C. reactor lower grid and can be moved slightly toward the open side and lowered

D. reactor lower grid and the assembly can be raised and jog can be used to reposition the assembly prior to lowering

**96** **ID: 1700737** **Points: 1.00**

Plant Conditions:

- Your shift has just assumed the watch in the Control Room.
- The Special Test Coordinator (a designee of the Senior Line Manager) has just commenced conducting a brief for Turbine Torsional Testing, which is scheduled to take place this shift.
- Due to one of the maintenance team members calling out sick, the testing group is short-handed.
- The Special Test Coordinator is briefing the following actions:
    - The Special Test Coordinator will assume the active role of the performer who called out sick.
    - Just in Time training is not required because the evolution was performed by the same team last time, so the members understand the actions required.
    - Station Management shall be present during the entire test.
    - The test is scheduled to take more than one shift and another brief will be required upon turnover.

In accordance with OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS, which one of the following is a true statement with regards to the actions that were briefed?

A. The Special Test Coordinator can not be an active performer of the test.

B. Station Management does not need to be present during the special test.

C. Only a Senior Line Manager can conduct the briefing for the special test.

D. Just in Time training must occur to ensure the test is understood by those involved.

**97** **ID: 1700778** **Points: 1.00**

Sequence of Events:

- An In-service Test (IST) performed on Nuclear River (NR) system pump NR-P-1A showed a flow rate of 6234 gpm, which is less than the minimum flow rate allowed by Technical Specification.
- Subsequent testing performed under a complex troubleshooting plan had shown:
  - Pump performance appeared to have leveled out at a lower flow rate, but still greater than the design minimum ASME value.
  - Engineering evaluations concluded that there would be no further degradation of flow rate.

After the troubleshooting plan is complete, NR-P-1A must be declared __(2)__.

A. Operable

B. Degraded

C. Inoperable

D. Unavailable

**98** **ID: 1700579** **Points: 1.00**

Both evaporators are out of service

Operations has initiated a liquid waste release permit in accordance with CY-TM-170-2001, RELEASING RADIOACTIVE LIQUID WASTE for the a WESCT.

The initial sample radioactivity was over 7E-7 uCi/ml (excluding tritium and noble gases).

Who one of the following is correct regarding the release of the WESCT?

A. The tank must be reprocessed and then released with Shift Manager approval.

B. The tank must be reprocessed and then released with Shift Manager and Chemistry Management approval.

C. The tank can be released with Shift Managment approval, ONLY.

D. The tank can be released with BOTH Shift Manager and Chemistry Management approval.

**99** **ID: 1685047** **Points: 1.00**

Plant Conditions:

- Reactor is at 100% power.
- A CODE WHITE has been declared by site security

EVENT:

- All communications between the ARO and control room have been lost
- CODE BLUE is declared and confirmed by site security
- CRS/URO continue with OP-TM-AOP-008, SECURITY THREAT / INTRUSION actions in the control room.

(1) When will the ARO start taking action after reporting to the 1E 4160V bus?
(2) What actions will the ARO take?

A. (1) When EF-P-2B is running
(2) Trip MU-P-1B, start MU-P-1A, and ensure seal injection is not lost to the reactor coolant pumps

B. (1) When EF-P-2B is running
(2) Take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to both OTSG's

C. (1) When the control room is breached
(2) Trip MU-P-1B, start MU-P-1A, and ensure seal injection is not lost to the reactor coolant pumps

D. (1) When the control room is breached
(2) Take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to both OTSG's

**100** **ID: 1700400** **Points: 1.00**

Plant Conditions:

- Reactor is operating at 100% power, with ICS in full automatic.
- You are the on-shift Control Room Supervisor.
- You have been relieved by the Shift Manager and are returning to the Control Room from the Operations Office Building (OOB).

EVENT:

Just as you pass through the Control Room entrance door:

- One-half of the overhead lights go out, and do NOT come back on.
- You observe the following MAPs actuate:
  - B-1-1, 4KV ES FDR BKR TRIP.
  - B-2-1, 4KV ES BUS UV/OV.
  - B-1-2, 4KV ES MOTOR TRIP.
  - F-1-5, RCP SEAL TOT INJECT FLOW HI/LO.

Based on these conditions, identify the ONE selection below that describes:

(1) The requirements for you to assume a management role in response to the upset, and
(2) The procedure that must be implemented to perform actions that are most critical to mitigation of the event.

A. (1) Obtain a brief update from the shift manager PRIOR to directing team activities.
(2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS.

B. (1) Obtain a brief update from the shift manager PRIOR to directing team activities.
(2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP.

C. (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.
(2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS.

D. (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.
(2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP.

# EXAMINATION ANSWER KEY



**1** **ID: 1685138** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Reactor Trip due to low condenser vacuum
- Condenser vacuum is 22 in Hg mercury and slowly lowering
- OTSG 'A' is at 98% in the operating range and slowly rising
- OTSG 'B' is at 25" in the startup range and steady

What actions will the operating crew take to stabilize the plant?

A. 1) Trip 'A' Main Feed Pump
   2) Stabilize OTSG pressure with the MS-V-3's (Turbine Bypass Valves)

B. 1) Trip both Main Feed Pumps
   2) Stabilize OTSG pressure with the MS-V-4's (Atmospheric Dump Valves)

C. 1) Trip 'A' Main Feed Pump
   2) Stabilize OTSG pressure with the MS-V-4's (Atmospheric Dump Valves)

D. 1) Trip both Main Feed Pumps
   2) Stabilize OTSG pressure with the MS-V-3's (Turbine Bypass Valves)

Answer: B

**Answer Explanation**

Explanation: To answer this question correctly the examinee must know: (1) on a low condenser vacuum, the MS-V-3's close and latch shut (2) at that point the MS-V-4's become the controlling valves to stabilize OTSG pressure at the MS-V-3 setpoint (3) the step to stabilize pressure is Step 3.10, which sends the operator to OP-TM-EOP-010, Guide 6 which directs the operator to MAINTAIN OTSG pressures below 1020 psig (4) due to the 'A' OTSG being greater than 97.5%, both of the Main Feed Pumps must be tripped, IAW OP-TM-EOP-001, step 3.6 RNO verifying BOTH OTSG levels below 97.5 % in the operating range, the RNO directs the operator ENSURE FW-P-1A and FW-P-1B (the Man Feed Pumps) are tripped.

| | | |
|---|---|---|
| A. | 1) Trip 'A' Main Feed Pump<br>2) Stabilize OTSG pressure with the MS-V-3's (Turbine Bypass Valves) | Plausible because the 'A' OTSG is the only OTSG with a high level, but due to the Main Feedwater System being cross-connected, both Main Feedwater Pumps must be tripped. In addition, if the examinee does not know that a main condenser vacuum < 23 in HG will close and latch the MS-V-3's (Turbine Bypass Valves) the examinee could choose this answer. |
| B. | 1) Trip both Main Feed Pumps<br>2) Stabilize OTSG pressure with the MS-V-4's (Atmospheric Dump Valves) | Correct Answer |
| C. | 1) Trip 'A' Main Feed Pump<br>2) Stabilize OTSG pressure with the MS-V-4's (Atmospheric Dump Valves) | Plausible because the 'A' OTSG is the only OTSG with a high level, but due to the Main Feedwater System being cross-connected, both Main Feedwater Pumps must be tripped. |
| D. | 1) Trip both Main Feed Pumps<br>2) Stabilize OTSG pressure with the MS-V-3's (Turbine Bypass Valves) | If the examinee does not know that a main condenser vacuum < 23 in HG will close and latch the MS-V-3's (Turbine Bypass Valves) the examinee could choose this answer. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 1 | |
| Group # | 1 | |
| K/A # | E02 | 2.1.20 |
| Importance Rating | 4.6 | 4.6 |

K/A: Ability to interpret and execute procedure steps.

Proposed Question: RO Question 1

Technical Reference(s): OP-TM-EOP-001, Rev 16 OP-TM-411-000, Rev 21

OP-TM-EOP-010, Rev 19

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP DBIG PCO-1

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

K/A Match: This question matches the KA because the examinee will have to interpret that the 'A' OTSG level is high, and that the procedurally driven action to execute is to trip both Main Feedpumps.
In addition, Guide 6, which is a required for plant stabilization, is required. The examinee will have to diagnose that the MS-V-3's (Turbine Bypass Valves) are closed and latched, and the the MS-V-4's (Atmospheric Dump Valves) are required to be utilized to stabilize the plant.

High Cog: This question is High Cog because the examinee must understand that when condenser vacuum lowers to < 22 in Hg, the plant response is that of cooldown to atmosphere and not the condenser. The MS-V-3 valves will latch closed, and transfer their setpoint to the MS-V-4 valves.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**2** **ID: 1737142** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- A Small Break LOCA has occurred inside containment.
- The leak is estimated at 25 gpm.
- Reactor Building pressure is 0.5 psig and rising at approximately 0.1 psig every two hours.
- Reactor Building temperature above 320' elevation is 120°F and rising at approximately 2°F every hour.

OTSG Startup level indicates __(1)__ than actual based on elevated Reactor Building __(2)__.

A. (1) lower
(2) pressure

B. (1) higher
(2) pressure

C. (1) lower
(2) temperature

D. (1) higher
(2) temperature

Answer: D

**Answer Explanation**

<<Explanation: To answer this question correctly the examinee must know: (1) The Startup Range Reference legs for OTSG level control are closed cell, wet reference legs; (2) Because of this, Reactor Building pressure will have no effect on the Startup Level indication, but Reactor Building temperature will; (3) As Reactor Building temperature rises, the water in the reference leg will become less dense; (4) When the water becomes less dense, the pressure on the High Pressure side of the D/P cell will lower, which lowers the D/P felt by the system; (5) The lower D/P results in a HIGHER than actual level.

| | | |
|---|---|---|
| A. | (1) lower<br>(2) pressure | INCORRECT: Reactor Building pressure will have no effect on the Startup Range level indication. Plausible because Reactor Building pressure is a parameter that is rising in the above scenario. |
| B. | (1) higher<br>(2) pressure | INCORRECT: Reactor Building pressure will have no effect on the Startup Range level indication. Plausible because Reactor Building pressure is a parameter that is rising in the above scenario. |
| C. | (1) lower<br>(2) temperature | INCORRECT: Indicated level will appear higher than actual level. Plausible because the D/P cell fells a lower pressure on the High Pressure side. |
| D. | (1) higher<br>(2) temperature | CORRECT: See above. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 009 | EK2.03 |
| | Importance Rating | 3.0 | 3.3 |

K/A: Small Break LOCA: Knowledge of the interrelationship between the small break LOCA and the following: S/Gs

Proposed Question: RO Question #2

Technical Reference(s): TQ-TM-104-411-C001, Rev 8 OS-24, Rev 28

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-2

Question Source: Bank # 375108

Modified Bank #

New

Question History: Sim Exam 5 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

| | | | |
|---|---|---|---|
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | 7 | |
| | 55.43 | | |

Comments:

KA Match; This question matches the KA because the examinee must determine the level error in an OTSG caused by a small break LOCA.

High Cog: The examinee must analyze the effects of a temperature rise on the OTSG level detectors.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**3** **ID: 1685156** **Points: 1.00**

Plant Conditions:

- Reactor is a 70% power
- All ICS stations are in HAND

EVENT:

- RC-P-1B trips

Tcold on the 'A' RCS loop will _______ and the operator will have to _______ 'A' feedwater flow to maintain a $0^0$F $\Delta$Tc.

A. rise
raise

B. rise
lower

C. lower
lower

D. lower
raise

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee will have to know: (1) RC-P-1B is in the 'A' RCS loop (2) because ALL ICS stations are in hand, the plant will respond to the loss of 'A' loop RCS flow, but the control system will keep the other parameters (i.e. Feedwater flow, Control Rod Position, etc) steady, (3) $\Delta$Tc is calculated by taking Tcold A - Tcold B, therefore with the loss of flow in the 'A' loop, the $\Delta$Tc will become negative, (4) to make the $\Delta$Tc become less negative, the operator will have to lower on 'A' feedwater flow.

| | | |
|---|---|---|
| A. | rise<br>raise | This is plausible if a reactor coolant pump in the 'B' loop were secured, and the question still asked how to make $\Delta$Tc by controlling the 'A' feedwater loop. |
| B. | rise<br>lower | This is plausible if the examinee misunderstands the direction Tcold will go when an 'A' loop Reactor Coolant Pump is secured. |
| C. | lower<br>lower | Correct Answer |
| D. | lower<br>raise | This is plausible if the student does not understand how to correct the $\Delta$Tc situation by with 'A' feedwater flow. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 1 | |
| Group # | 1 | |
| K/A # | 015 | AA1.09 |
| Importance Rating | 3.1 | 3.2 |

K/A: Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Pump Malfunction (Loss of RC Flow): RCS temperature detection system.

Proposed Question: RO Question #3

Technical Reference(s): TQ-TM-104-621-C001, Rev 10

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-GLO-11

Question Source: Bank # 858178

Modified Bank #

New

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

| | | |
|---|---|---|
| | Comprehension or Analysis | X |
| 10 CFR Part 55 Content: | 55.41 | 7 |
| | 55.43 | |

Comments: (KA Match and why high cog)

K/A Match: This matches the K/A because the examinee will have to describe what they expect RCS Tcold to do upon securing a reactor cooling pump.. In addition, the question requires the operator to know how to operate the FW control station associated with level in the OTSG in which the reactor coolant pump was secured. The RCS temperature detection subsystem in the question is the $\Delta$Tc ICS circuit which monitors the 'A' and 'B' RCS cold leg temperatures. When in automatic, this ICS circuit uses FW demand and $\Delta$Tc to maintain a near zero cold leg temperature. If the system were in automatic, when a reactor coolant pump is secured, the system will automatically re-ratio feedwater to maintain a near zero $\Delta$Tc.

High Cog: This question is high cog because the examinee will have understand the what the impact of securing a reactor coolant pump has to analyze how the RCS parameters will change, and how to manipulate controls to return those parameters to normal.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**4** **ID: 1685199** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- MU-P-1B trips on overload.
- MAP G-2-5 PZR LEVEL HI-LO annunciator has alarmed.
- Pressurizer level is 196 inches and slowly lowering.

Which one of the following identifies actions that the operator must take while restoring a Makeup Pump to operation?

A. De-energize all Pzr Heaters; AND
CLOSE MU-V-33A-D, Control Bleed Off Valves.

B. CLOSE MU-V-3, Letdown Isolation Valve; AND
CLOSE MU-V-33A-D, Control Bleed Off Valves.

C. De-energize all Pzr Heaters; AND
Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to HAND and CLOSE the valves.

D. CLOSE MU-V-3, Letdown Isolation Valve; AND
Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to HAND and CLOSE the valves.

Answer: D

**Answer Explanation**

<<Explanation: To answer this question correctly the examinee must know: (1) Since there is no makeup flow, the inventory will begin to lower. Pressurizer level will lower due to the normal letdown and CBO flow of the the reactor coolant pumps. (2) The crew will enter OP-TM-AOP-041 for a loss of Seal Injection (due to the MU-P-1B trip), which directs MU-V-32 to be closed in step 3.3 and MU-V-3 and MU-V-17 to be closed in step 3.5 RNO.

| | | |
|---|---|---|
| A. | De-energize all Pzr Heaters; AND CLOSE MU-V-33A-D,Control Bleed Off Valves. | This is plausible because the operator may incorrectly believe that the heaters will need to be deenergized due to low pressurizer level; and because the operator may incorrectly believe that procedures direct the closure of MU-V-33A-D to save RCS inventory from being lost. |
| B. | CLOSE MU-V-3, Letdown Isolation Valve; AND CLOSE MU-V-33A-D, Control Bleed Off Valves. | This is plausible because the operator may incorrectly believe that procedures direct the closure of MU-V-33A-D to save RCS inventory from being lost. Closing MU-V-3 would save inventory, but incorrect due to MU-V-33's are not required to be closed. |
| C. | De-energize all Pzr Heaters; AND Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to HAND and CLOSE the valves. | This is plausible because the operator may incorrectly believe that that the heaters will need to be deenergized due to low pressurizer level. According to OP-TM-MAP-G0305, at 80 inches in the Pzr the Pzr Heaters will automatically trip. However, this level has not been reached, and the need to de-energize the Pzr heaters is not identified in either EOP-10 Guide 9, or AOP-041. |
| D. | CLOSE MU-V-3, Letdown Isolation Valve; AND Take MU-V-17, Makeup Flow Control Valve, and MU-V-32, Seal Injection Valve, to HAND and CLOSE the valves. | Correct Answer |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 1 | |
| Group # | 1 | |
| K/A # | 022 | AK1.03 |
| Importance Rating | 3.0 | 3.4 |

K/A: Knowledge of the operational implications of the following concepts as they apply to the Loss of Reactor Coolant Makeup: Relationship between charging flow and PZR level.

Proposed Question: RO Question #4

Technical Reference(s): OP-TM-AOP-041, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-041-PCO-1

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question Source: | Bank # | 719461 | |
| | Modified Bank # | | |
| | New | | |
| Question History: | Sim Exam 7 | Last NRC Exam: | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | 8 | |
| | 55.43 | | |

Comments:

K/A Match: The KA is matched because the operator must demonstrate knowledge (i.e. what actions must be taken) of the operational implications (i.e. RCS Inventory is being lost, and action that are needed to start MU Pump restart) of the concept of the relationship between charging flow and Pzr level as it applies to the Loss of Reactor Coolant Pump Makeup. Due to the loss of the charging flow, pressurizer level will lower, so letdown is isolated by closing MU-V-3 to minimize the less of inventory from the pressurizer. MU-V-17(Pressurizer level control valve) and MU-V-32 (Seal Injection Valve) are closed in hand for a controlled restoration of makeup and seal injection upon the restoration of a makeup pump.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**5** **ID: 1737146** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- All available Nuclear River water pumps trip, and cannot be restarted.
- OP-TM-AOP-031, LOSS OF NUCLEAR SERVICES COMPONENT COOLING is entered.

Which one of the following describes the strategy and basis for Reactor Coolant Pump (RCP) operation?

A. The RCPs must be secured due to the loss of labyrinth cooling to each RCP.

B. The RCPs must be secured to eliminate heat load from the NSCCW System.

C. The RCPs may remain in operation provided seal injection flow is greater than 22 gpm.

D. The RCPs may remain in operation provided the SR-NR cross tie is performed prior to reaching the HI-2 alarm setpoint for the RCPs.

Answer: B

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) After the Reactor is tripped in OP-TM-AOP-031, the next step the operating crew must do is trip all four Reactor Coolant Pumps; (2) According the the basis, this is to eliminate heat load from the Nuclear Services system.

| | | |
|---|---|---|
| A. | The RCPs must be secured due to the loss of labyrinth cooling to each RCP. | Incorrect: Plausible because Intermediate Closed Cooling Water (which is cooled by Nuclear River Water) does cool the thermal barrier on the reactor coolant pumps. Incorrect because the seal injection cools the labyrinth seal. |
| B. | The RCPs must be secured to eliminate heat load from the NSCCW System. | Correct Answer: See above. |
| C. | The RCPs may remain in operation provided seal injection flow is greater than 22 gpm. | Incorrect: Plausible because Intermediate Closed Cooling does cool the thermal barrier of the Reactor Coolant Pump, and there is an interlock with ICCW and SI. Incorrect because the Reactor Coolant Pumps must be secured because of the lack of cooling to the RCP motors and the heat load on the NSCCW system. |
| D. | The RCPs may remain in operation provided the SR-NR cross tie is performed prior to reaching the HI-2 alarm setpoint for the RCPs. | INCORRECT: Plausible because a cross connect does exist between Secondary River and Nuclear River water. The systems are cross connected in OP-TM-AOP-031. Incorrect because the Reactor Coolant Pumps are a major heat load on the system and must be secured prior to cross connecting. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 026 | AK3.03 |
| | Importance Rating | 4.0 | 4.2 |

K/A: Loss of Component Cooling Water: Knowledge of the reasons for the following responses as they apply to Loss of Component Cooling Water: Guidance actions contained in EOP for Loss of CCW.

Proposed Question: RO Question #5

Technical Reference(s): OP-TM-AOP-031, Rev 6

OP-TM-AOP-031, Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-031-PCO-4

Question Source: Bank # 634661

Modified Bank #

New

| | | | |
|---|---|---|---|
| Question History: | Sim Exam 6 | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments:

K/A Match: This question matches the KA because the examinee must have knowledge of an action contained in the AOP for Loss of Nuclear Services Component Cooling.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**6** **ID: 1685231** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- The PORV fails partially open.
- Efforts to close the PORV Block valve have failed.

Current Plant Conditions:

- The plant is being shutdown.
- RCS Pressure is 2050 psig and slowly lowering.
- RCS Subcooling margin is 36°F.
- Pressurizer level is 250 inches and slowly rising.

In accordance with OP-TM-AOP-043, LOSS OF PRESSURIZER, assuming the plant does NOT go solid, what action must be taken regarding operation of MU-V-17, Pressurizer Level Control Valve?

Place MU-V-17 in hand and adjust makeup flow to maintain __(1)__

A. > 40°F SCM by squeezing the pressurizer bubble.

B. > 150 inches in the pressurizer to ensure adequate inventory on the subsequent RCS cooldown.

C. > 70°F SCM by squeezing the pressurizer bubble.

D. > 220 inches in the pressurizer to ensure adequate inventory on the subsequent RCS cooldown.

Answer: A

**Answer Explanation**

<<<<Explanation: To answer this question correctly, the examinee must know: (1) The plant is being shutdown because there is inadequate heater capacity to maintain RCS pressure; (2) To maintain RCS pressure high enough to complete the shutdown, the steam bubble in the pressurizer will have to be squeezed, and pressurizer level will go up; (3) With current plant conditions, there is adequate pressurizer level indication and the plant is not solid.

| | | |
|---|---|---|
| A. | > 40F SCM by squeezing the pressurizer bubble | CORRECT ANSWER |
| B. | > 150 inches in the pressurizer to ensure adequate inventory on the subsequent RCS cooldown. | INCORRECT ANSWER: Plausible because >150 inches is trip criterial in some EOP's. The RCS will be cooling down and Pressurizer Level will have to be made up to accordingly. |
| C. | > 70°F SCM by squeezing the pressurizer bubble. | INCORRECT ANSWER: Plausible because this is the action to take if the pressurizer goes solid. Incorrect because the question states that the pressurizer does NOT go solid. |
| D. | > 220 inches in the pressurizer to ensure adequate inventory on the subsequent RCS cooldown. | INCORRECT ANSWER: Plausible because 220 inches is the normal pressurizer level. The examinee could believe that level should be maintained at 220 inches during the cooldown. Incorrect because OP-TM-AOP-043 gives guidance to maintain SCM >40F in this scenario. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 1 | |
| Group # | 1 | |
| K/A # | 027 | AA2.12 |
| Importance Rating | 3.7 | 3.8 |

K/A: Ability to determine and interpret the following as they apply to Pressurizer Pressure Control Malfunctions: PZR Level

Proposed Question: RO Question #6

Technical Reference(s): OP-TM-AOP-043, Rev 6

Steam Tables

Proposed References to be provided to applicants during examination: Steam Tables

Learning Objective: AOP-043-PCO-4

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | 5 | |
| | 55.43 | | |

Comments:

K/A Match: This is a K/A match because the PORV is part of the Pressurizer Pressure Control system, in that it receives an input from a narrow range pressure detector and opens when a setpoint is reached. Due to the PORV being open far enough to effectively lower pressure beyond the capacity of the pressurizer heaters, a shutdown must commence. To effectively ensure we have time to shutdown before tripping on a low pressure signal, the Pressurizer Level Control Valve, MU-V-17 is taken to HAND, and flow is raised to squeeze the bubble in the pressurizer. The operator will control pressurizer level to maintain a subcooling margin > 40F which the examinee can determine with the steam tables.

High Cog: This question is high cog because the examinee will have to analyze the current plant parameters and determine which subcooling margin guidance is applicable.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**7** **ID: 1685342** **Points: 1.00**

Event:

- Loss of offsite power.
- Emergency Diesel Generator, EG-Y-1A fails to start.
- Emergency Diesel Generator, EG-Y-1B trips after starting.

Based on these conditions, which of the following control room extension controls or instruments remain operable?

A. RM-A-2 Sample Pump.

B. Motor Driven Fire Pump FS-P-2.

C. 1SB-D2, 1B Auxiliary Transformer feeder breaker to 1D 4KV Switchgear.

D. RR-PI-225, RB Emergency Cooling Outlet Cooler 1B pressure instrument.

Answer: C

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) the DC electrical system is used to supply ES breaker control power and protective relaying.

| | | |
|---|---|---|
| A. | RM-A-2 Sample Pump. | Plausible because this is safety related equipment, but RM-A-2 sample pump is powered from 1A ES 480V MCC and uses 120V control power transformer housed in the breaker cubicle. |
| B. | Motor Driven Fire Pump FS-P-2. | Plausible because it is safety related equipment, but employs AC control power. |
| C. | 1SB-D2, 1B Auxiliary Transformer feeder breaker to 1D 4KV Switchgear. | Correct Answer. |
| D. | RR-PI-225, RB Emergency Cooling Outlet Cooler 1B pressure instrument. | Plausible because this is is an ES pressure indicator, but due to lack of power to TRB none of the RB Emergency Cooler outlet pressure instruments are available. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 055 | EA2.04 |
| | Importance Rating | 3.7 | |

K/A: Ability to determine or interpret the following as they apply to a Station Blackout: Instruments and controls operable with only dc battery power available.

Proposed Question: RO Question #7

Technical Reference(s): 1107-5, Rev 151

TQ-TM-104-740-C001, REV 7

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-020-PCO-2

Question Source: Bank #

Modified Bank # 363906

New

Question History: Sim Exam 8 Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

K/A Match: This matches the K/A because the examinee must know what controls are DC power (which would make them available during a station blackout) in the control room to answer the question correctly. The question choices are a mixture of instrument and controls.

Modified Bank Question: This question was modified by changing the failure of EG-Y-B to trip after starting, and replacing MU-V-16C with an instrument (RR-PI-225). This replacement was performed to better match the K/A.
Event:

- Loss of offsite power.
- Emergency Diesel Generators EG-Y-1A and EG-Y-1B both failed to start.

Based on these conditions, identify the ONE statement below that describes control room extension controls that remain operable.

A. RM-A-2 Sample Pump.

B. Motor Driven Fire Pump FS-P-2.

C. High Pressure Injection valve MU-V-16C.

D. 1SB-D2, 1B Auxiliary Transformer feeder breaker to 1D 4KV Switchgear.

Answer: D

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**8** **ID: 1685400** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- Station Blackout Diesel Generator, EG-Y-4 is unavailable due to an outage.

EVENT:

- Loss of Offsite power.
- Emergency Diesel Generator, EG-Y-1B has started and is powering 'E' 4kV Bus.
- Emergency Diesel Generator, EG-Y-1A has tripped on overspeed.

What action per OP-TM-AOP-020, Loss of Station Power, will be taken to prevent the loss of plant control and provide additional time to restore AC power?

A. Secure Emergency DC Seal Oil Pump GN-P-2.

B. Initiate OP-TM-732-901, Energize 1P 480V Bus using the ES Bus Cross Tie.

C. Initiate OP-TM-732-902, Energize 1S 480V Bus using the ES Bus Cross Tie.

D. Ensure Turning Gear Oil Pump LO-P-5 is running, then secure Emergency Bearing Oil Pump LO-P-6.

Answer: B

**Answer Explanation**

To answer this question, the examinee must know (1) EG-Y-1A powers the 'D' 4kV bus (2) since EG-Y-1A and EG-Y-4 are unavailable, the 'D' 4kV bus has no power supply (3) 1P 480V Bus is downstream of the 'D' 4kV bus and has no power (4) the ES busses can be crosstied with the reactor is not critical

A. "Secure Emergency DC Seal Oil Pump." - Incorrect - Plausible since this equipment is identified in step 3.15 as "verify operating" in this AOP procedure not securing the pump.

B. "Initiate OP-TM-732-901, Energize 1P 480V Bus using the ES Bus Cross Tie.." - Correct - This is step 3.21. This energizes the battery chargers for the "A" station battery. In accordance with OP-TM-AOP-0201, Loss of Station Power basis document, *if AC is only available to one train, the 480V ES busses will be crosstied and large DC loads will be minimized. This will allow for additional time to restore an AC power source. The procedures to crosstie the ES busses can be performed without jeopardizing the reliability of the available AC supply and will prevent the loss of plant control caused by loss of DC power and two vital busses.*

C. "Initiate OP-TM-732-902, Energize 1S 480V Bus using the ES Bus Cross Tie."- Correct Answer - Plausible because this is a step in the procedure, however it is for when the 1E 4kV is de-energized not the situation given in the stem.

D. "Ensure Turning Gear Oil Pump LO-P-5 is running, then secure Emergency Bearing Oil Pump LO-P-6." - Incorrect - Plausible, since LO-P-6 is powered from "A" DC and will be running. LO-P-5 is powered from 1P 480V and will not be running. Step 3.13 directs the examinee to ensure LO-P-5 or LO-P-6 is running. Once that is verified, the examinee would move to the next step with no action taken.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 056 | AK3.02 |
| | Importance Rating | 4.4 | |

K/A: Knowledge of the reasons for the responses as they apply to the Loss of Offsite Power: Actions contained in the EOP for loss of offsite power.

Proposed Question: RO question #8

Technical Reference(s): OP-TM-AOP-020, Rev 24

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP020-PCO-4

Question Source: Bank #

Modified Bank # 1244855

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

New

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments: This is a KA match because contains actions in our loss of offsite power AOP. The loss of offsite power AOP for TMI is essentially an EOP in broad terms. The loss of offsite power EOP for TMI deals the an extended loss of offsite power.

Original Question:

Plant Conditions:

- Plant is at 100% power
- EG-Y-4 is unavailable due to an outage.

Event:

- Loss of Offsite power.
- Emergency Diesel Generator, EG-Y-1A has started and is powering 'D' 4kV Bus
- EG-Y-1B has tripped on overspeed.

What action per OP-TM-AOP-020, Loss of Station Power, will be taken to prevent the loss of plant control and provide additional time to restore AC power?

A. Secure Emergency DC Seal Oil Pump GN-P-2.

B. Initiate OP-TM-732-901 Energize 1P 480V Bus using the ES Bus Cross Tie.

C. Initiate OP-TM-732-902 Energize 1S 480V Bus using the ES Bus Cross Tie.

D. Ensure Turning Gear Oil Pump LO-P-5 is running, then secure Emergency Bearing Oil Pump LO-P-6.

Answer: C

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**9** **ID: 1685632** **Points: 1.00**

Plant Conditions:

- Plant Cooldown is in progress per 1102-11, Plant Cooldown.
- Current Cooldown rate is 85°F/HR.
- Current RCS temperature is 359°F and RCS Pressure is 600 psig.

EVENT:

- Channel 1 of B train ES indicates actuated.
- NI-5 loses indication on console center.
- MU-V-2A/2B Letdown Cooler Outlet Valves close.

(1) Which one of the following describes the action required if RCS temperature lowers below 313°F?
(2) What is the reason for the action taken?

A. (1) Open RC-RV-2 (PORV) until RCS pressure is below 540 psig.
(2) The NDTT interlock has lost power.

B. (1) Open RC-V-1 (RCS Spray Valve) until RCS pressure is below 540 psig.
(2) The PORV and the NDTT interlock have lost power.

C. (1) Open RC-RV-2 (PORV) until RCS pressure is below 540 psig.
(2) RC-V-1 and the NDTT Interlock have lost power.

D. (1) Open RC-V-1 (RCS Spray Valve) until RCS pressure is below 540 psig.
(2) The NDTT interlock has lost power.

Answer: A

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) Channel 1 of B train ES appearing actuated, NI-5 losing indication on console center, and MU-V-2A/2B letdown cooler outlet valves closing are all indications that Vital Bus A is lost; (2) the reason the stem indicates that on Channel 1 of B train ES appears actuated vice Channel 1 of A and B train ES appears actuated is because the A train ES lights are powered from Vital Bus A, and this information would not be apparent to the operator until the ES Status Light Power Supply Select Switch is positioned to the BUS-B position; (3) The operating crew would enter OP-TM-AOP-015, LOSS OF VBA, which would direct the operating crew to open RC-RV-2 if RCS pressure were > 590 psig when RCS temperature is < 313°F due to NDTT interlock not being operable.

| | | |
|---|---|---|
| A. | (1) Open RC-RV-2 (PORV) until RCS pressure is below 540 psig.<br>(2) The NDTT interlock has lost power. | Correct Answer |
| B. | (1) Open RC-V-1 (RCS Spray Valve) until RCS pressure is below 540 psig.<br>(2) The PORV and the NDTT interlock have lost power. | INCORRECT: Plausible because RC-V-1 (RCS Spray Valve) is usually used to lower RCS pressure. At this temperature and pressure, the PORV is used due to NDTT concerns. In addition, the PORV did not lose power. |
| C. | 1) Open RC-RV-2 (PORV) until RCS pressure is below 540 psig.<br>(2) RC-V-1 and the NDTT Interlock has lost power. | INCORRECT: Plausible because RC-RV-2 is what the operators would use to lower pressure in this case. Incorrect because RC-V-1 did not lose power (1A ES MCC). |
| D. | (1) Open RC-V-1 (RCS Spray Valve) until RCS pressure is below 540 psig.<br>(2) The NDTT interlock has lost power. | Plausible because RC-V-1 (RCS Spray Valve) is usually used to lower RCS pressure. At this temperature and pressure, the PORV is used due to NDTT concerns. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 057 | AK3.01 |
| | Importance Rating | 4.1 | 4.1 |

K/A: Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus.

Proposed Question: RO Question #9

Technical Reference(s): OP-TM-AOP-015, Rev 10

OP-TM-AOP-0151, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-015-PCO-4

Question Source: Bank #

Modified Bank # 606540

New

Question History: Sim Exam 5 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments: (KA Match and why high cog)

K/A Match: This matches the KA because the examinee can determine that the PORV is used to lower pressure due to NDTT concerns (vice the spray valve). Loss of a vital bus is an AOP at this facility. The examinee must determine which AOP is entered and the action within.

High Cog: This question is high cog because the examinee will have to analyze plant conditions and determine which vital bus was lost.

Plant Conditions:

- Plant Cooldown is in progress per 1102-11, Plant Cooldown.
- Current Cooldown rate is 85°F/HR.
- Current RCS temperature is 359°F and RCS Pressure is 600 psig.

Sequence of Events:

- "A" inverter fails and VBA is lost.
- OP-TM-AOP-015, Loss of VBA is entered.
- 15 minutes into the casualty the Unit Reactor Operator reports "RCS Temperature is 312°F and RCS Pressure is 595 psig.

Based on the report, ___(1)___ should be opened and should remain open until RCS pressure is less than the procedurally required value of ___(2)___ psig.

A. (1) RC-RV-2 (PORV)
(2) 540

B. (1) RC-RV-2 (PORV)
(2) 500

C. (1) RC-V-1 (RCS Spray Valve)
(2) 540

D. (1) RC-V-1 (RCS Spray Valve)
(2) 500

Answer: A

**10** **ID: 1699640** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- Operating crew is performing Emergency Diesel Generator, EG-Y-1A, 24 hour endurance run in accordance with 1107-3.
- Currently EG-Y-1A is operating in parallel with the grid.

EVENT:

The following alarms are received simultaneously:

- Battery 1A Discharging, A-1-7.
- Battery Charger 1A/1C/1E Trouble, A-2-7.
- Inverter 1A/1C/1E System Trouble, A-3-7.
- CRDM Breaker Test Trouble, PRF 1-1-1.
- 230 KVolt Substation Trouble, NN-3-1.
- 7 KVolt Bus Trouble, AA-3-2.
- 4 KVolt BOP Bus Trouble, AA-3-3.
- 480 Volt BOP Bus Trouble, AA-3-5.

In order to protect EG-Y-1A the operating crew must:

G1-02 = EG-Y-1A Diesel output breaker to 1D 4160V Bus
EG-V-15A = Air Start Header Isolation Valve to EG-Y-15A

A. Close EG-V-15A, only.

B. Close EG-V-15A and trip EG-Y-1A fuel rack.

C. Open breaker G1-02 from the control room and close EG-V-15A.

D. Open breaker G1-02 using the local OPEN pushbutton and close EG-V-15A.

Answer: D

| **Answer Explanation** |
| --- |

<<Explanation: To answer this question correctly, the examinee must know: (1) With the alarms present, there is a complete loss of 'A' DC; (2) when DC power is lost, EG-Y-1A loses its excitation field, in addition the ability to operate breakers associated with the 1D 4160V bus is lost (3) when DC is lost to EG-Y-1A with the generator breaker closed (parallel with the grid) the diesel will act as an inductive load and damage will occur (TQ-TM-104-861-COO1); (4) in addition, the air start solenoid will open which results in the air start distributor staying engaged, resulting in damage to the distributor.

| | | |
|---|---|---|
| A. | Close EG-V-15A, only. | Plausible because if the examinee determines that a complete loss of 'A' DC occurred without regard to the current status of the diesel (which is parallel to the grid), then this would be the correct answer. |
| B. | Close EG-V-15A and trip EG-Y-1A fuel rack. | Plausible because these actions are in OP-TM-AOP-023, which would be entered when 'A' DC was lost. Tripping the fuel racks is to shutdown the diesel, not for diesel protection. In addition, performing these steps would not open the diesel output breaker G1-02 which is causing the diesel to become an inductive motor. |
| C. | Open breaker G1-02 from the control room and close EG-V-15A. | Plausible because these actions would protect the diesel. Incorrect because G1-02 must be opened locally at the breaker because 'A' DC control power was lost, and the control room extension control will not work. |
| D. | Open breaker G1-02 using the local OPEN pushbutton and close EG-V-15A. | Correct answer - OP-TM-AOP-023 (step 4.15) directs the crew to close EG-V-15A. 1107-3 (Limit and Precaution 2.1.2) directs the breaker to be opened locally with the manual plunger. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 058 | AK3.01 |
| | Importance Rating | 3.4 | 3.7 |

K/A: Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of dc control power by the D/Gs.

Proposed Question: RO Question #10

Technical Reference(s): 1107-3, Rev 146

OP-TM-AOP-023, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-GLO-11

Question Source: Bank #

Modified Bank #

New X

| | | | |
|---|---|---|---|
| Question History: N/A | | Last NRC Exam: N/A | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.5 | |
| | 55.43 | | |

Comments:

K/A Match: The question matches the K/A because the examinee will have to know what the effect of a loss of DC has on a running/paralleled diesel generator. The examinee will have to know that it effects the control power to the diesel generator breaker, in addition to the effects on the starting air solenoid.

High Cog: The examinee will have to analyze the alarms present, and the present condition of the diesel (paralleled with the grid) to determine the steps needed to protect the diesel generator.

**11** **ID: 1699734** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- NR-P-1C is out of service for maintenance.
- NR-P-1A is ES selected and running on 1R 480v bus.
- NR-P-1B is ES selected and running on 1T 480v bus.

EVENT:

- Loss of Offsite Power occurs.
- RCS pressure lowered to 1575 psig and is now rising slowly.
- RB pressure is 1.5 psig and relatively steady.

EVENT + 10 seconds:

- The Diesel Generators have started and are powering the ES Buses.

Which one of the following identifies the Nuclear River Water pump(s) that are running, if any, 1 minute after the re-energization of the ES Buses?

A. NR-P-1A ONLY.

B. NR-P-1B ONLY.

C. NR-P-1A and NR-P-1B.

D. No NR pump is running.

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) When an loss of of offsite power occurs, all nuclear river water pumps shut off; (2) At an RCS pressure less than 1600 psig, and ES signal is generated; (3) When the diesels start and power the ES busses, and with an ES signal present, the remaining two Nuclear River Water pumps will start on Block 3.

| | | |
|---|---|---|
| A. | NR-P-1A ONLY. | Plausible if the student believes a pump will start on standby. NR-P-1A would receive a start signal first if NR-P-1C tripped. |
| B. | NR-P-1B ONLY. | Plausible because had the plant been in a normal lineup with NR-P-1A and NR-P-1C ES selected and running, then NR-P-1B would be the only running Nuclear River Water Pump after a loss of offsite power event (with no ES). |
| C. | NR-P-1A and NR-P-1B. | Correct Answer |
| D. | No NR pump is running. | Plausible because in this lineup, if no ES signal were present, then no NR pumps would be running. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 1 | |
| Group # | 1 | |
| K/A # | 062 | AK302 |
| Importance Rating | 3.6 | 3.9 |

K/A: Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of ESFAS.

Proposed Question: RO Question #11

Technical Reference(s): TQ-TM-104-531-C001, Rev 9
TQ-TM-104-740-C001, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 740-GLO-10

Question Source: Bank # 298156
Modified Bank #
New

Question History: System Exam 6 Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

K/A Match: This matches the K/A because the examinee has to understand what happens to the Nuclear River Water pumps on an Loss of Offsite power in conjunction with an ES actuation signal. The Nuclear River Water pumps are part of the Nuclear Service Water system, they provide cooling to the Nuclear Service Water Coolers.

High Cog: The examinee will have to analyze the current plant conditions: Only 2 nuclear river water pumps are available, and they that they are ES selected, that we have an Loss of Offsite power and that RCS pressure is 1575 psig, and understand that when RCS pressure is below 1600 psig, that an ES signal is present. Both nuclear service river water pumps trip when offsite power is lost and, both start on Block 3 ES after power is restored.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**12** **ID: 1700681** **Points: 1.00**

<<REFERENCE PROVIDED

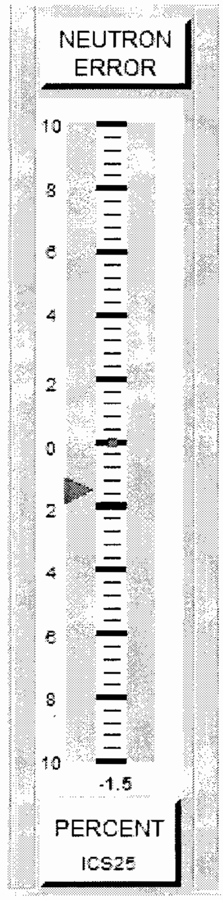
Plant Conditions

- Reactor power is 75% with ICS in auto.

EVENT:

- NI-5 power starts to slowly lower causing a SASS Mismatch.
- FW-V-17A, Main Feedwater Control Valve, starts to fail closed, causing a Feedwater to Reactor Cross Limit.
- Neutron Error is at -1.5% on console center.

Based on the above plant conditions, control rods will __(1)__ due to __(2)__.




A. (1) insert
(2) NI-5 failing

B. (1) insert
(2) Feedwater to Reactor Cross Limits

C. (1) withdraw
(2) NI-5 failing

D. (1) withdraw
(2) Feedwater to Reactor Cross Limits

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) NEGATIVE indicated neutron error results in control rod withdraw (Neutron Error = Reactor Demand - Actual Power); (2) NI-5 power drifting low causes a positive neutron error which is displayed on the console as a negative neutron error; (3) A SASS MISMATCH alarm prevents NI-5 from automatically swapping to the good instrument (NI-6); (4) NI-5 is the normally selected power range detector, with a SASS MISMATCH in, NI-5 will continue to be the controlling nuclear instrument, even though it is failing low; (5) The result is a situation where control rods will withdrawal to attempt to zero out neutron error, but as long as NI-5 continues to fail low, the negative neutron error will demand that control rod withdrawal; (6) Feedwater to Reactor Cross Limits cannot raise Reactor Power, therefore will not cause control rods to withdrawal.

| | | |
|---|---|---|
| A. | (1) insert<br>(2) NI-5 failing | INCORRECT: NI-5 failure LOW would create a neutron error to the CRD System, while this error is positive (with respect to ICS actual power vs demanded power) the operator would see an inverted signal in which neutron error would appear negative and demand a control rod withdrawal. Rods inserting is plausible because this negative indication may cause the examiner to believe the control rods should be inserting. |
| B. | (1) insert<br>(2) Feedwater to Reactor Cross Limits | INCORRECT: Control Rods would withdraw. The examinee may reason control rods would be inserting. Rods inserting is plausible because this negative indication may cause the examiner to believe the control rods should be inserting. |
| C. | (1) withdraw<br>(2) NI-5 failing | CORRECT Answer: Indicated Neutron Error on console indication is reversed polarity to the actual neutron error going to the CRD System AND FW Cross Limits can only LOWER Reactor power, i.e. drive Control Rods IN. |
| D. | (1) withdrawal<br>(2) Feedwater to Reactor Cross Limits | INCORRECT: Feedwater Cross Limits cannot raise Reactor Power, however with the indicated neutron error the operator may reason the Cross Limit is inducing this error. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 001 | AA2.05 |
| | Importance Rating | 4.4 | 4.6 |

K/A: Continuous Rod Withdrawal Ability to determine and interpret the following as they apply to Continuous Rod Withdrawal: Uncontrolled rod withdrawal, from available indication

Proposed Question: RO Question #12

Technical Reference(s): ICS Analog Print: D553732

TQ-TM-104-621-C001, Rev 10

Proposed References to be provided to applicants during examination: D553732

Learning Objective: 621-GLO-11

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments:

K/A Match: The conditions given in the question result in a Continuous Rod Withdrawal: indication of NI-5 failing low and Neutron Error. The operator must understand the indication and the final result of the multiple indications.

High Cog: The operator must understand the relationship between neutron error and the signal sent to the CRD System. While a negative error may imply Control Rods would insert, the ICS Signal would be positive and rods would withdraw. With a Feedwater to Reactor Cross Limit in effect the operator must understand the ICS does not allow this cross limit to raise reactor demand, only allows reactor demand to be lowered.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**13** **ID: 1741874** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- Pressurizer Level LT-1 is selected as the controlling channel.

EVENT:

- A leak develops on the reference leg of LT-1 causing pressurizer level to change 100 inches over the next 5 seconds.

Based on these conditions, indicated (LT-1) Pressurizer level has ___(1)___ and assuming no operator action ___(2)___.

A. (1) risen
(2) MU-V-17, Pressurizer Level Control Valve, will close

B. (1) risen
(2) the controlling Pressurizer Level Transmitter will SASS from LT-1 to LT-3

C. (1) lowered
(2) MU-V-17, Pressurizer Level Control Valve, will open

D. (1) lowered
(2) the controlling Pressurizer Level Transmitter will SASS from LT-1 to LT-3

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) The pressurizer uses a wet reference leg, which means a 0 DP when the pressurizer is full; (2) Normal Level setpoint is 220 inches; (3) As the reference leg loses water, the DP would lower which would be seen as a pressurizer level rise on the PZR Level indication; (4) Since indicated pressurizer level is rising, MU-V-17, Pressurizer Level Control valve, will close to maintain level at setpoint of 220 inches; (5) Actual pressurizer would lower due to letdown flow remaining constant and makeup flow lowering due to MU-V-17 closing.

| | | |
|---|---|---|
| A. | (1) risen<br>(2) MU-V-17, Pressurizer Level Control Valve, will close | **Correct.** Pressurizer level instruments RC-1- LT-1 and LT-3, are manually selected on RC-1 LR recorder on CC. The level transmitters use a wet reference leg such that a full pressurizer has 0" DP, and an empty pressurizer has 400" DP. Consequently, a leak in the reference leg will reduce the differential pressure between the reference and the variable leg. As the differential pressure is reduced, indicated level will rise. Pressurizer level is maintained at 220" when the plant is operating at 100% full power. The level is sensed and applied to a level control circuit. The level control circuit when in the automatic mode will control the position of the normal makeup valve (MU-V-17). If the level drops below 220" the circuit will open the normal makeup valve until the levels returns to normal, and vice-versa. Consequently, when the reference leg leak occurs, the indicated level will start to rise, causing MU-V-17 to close. Since there is less makeup flow, pressurizer level will actually lower. |
| B. | (1) risen<br>(2) the controlling Pressurizer Level Transmitter will SASS from LT-1 to LT-3 | **Incorrect.** This is plausible because the operator may misunderstand the Pressurizer Level detector operation. Although many Non Nuclear Instruments (NNI) use the Smart Automatic Signal Selector (SASS) pressurizer level transmitters is NOT one of those NNI's. To enhance plausibility, the pressurizer level changes at a rate which would actuate the SASS to select a new instrument. |
| C. | (1) lowered<br>(2) MU-V-17, Pressurizer Level Control Valve, will open | **Incorrect.** This is plausible because the operator may incorrectly believe that when the leak is sensed that the level control will cause indicated level to lower. This is incorrect because as stated above, level indication will rise. In addition, if indicated level were to lower, MU-V-17 would open up to attempt to maintain level. |
| D. | (1) lowered<br>(2) the controlling Pressurizer Level Transmitter will SASS from LT-1 to LT-3 | **Incorrect.** This is plausible because the operator may misunderstand the Pressurizer Level detector operation. Although many Non Nuclear Instruments (NNI) use the Smart Automatic Signal Selector (SASS) pressurizer level transmitters is NOT one of those NNI's. To enhance plausibility, the pressurizer level changes at a rate which would actuate the SASS to select a new instrument. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 028 | AK1.01 |
| | Importance Rating | 2.8 | 3.1 |

K/A: Knowledge of the operational implications of the following concepts as they apply to Pressurizer Level Control Malfunctions: PZR reference leg leaks.

Proposed Question: RO Question #13

Technical Reference(s): TQ-TM-104-624-C001, Rev 4

TQ-TM-104-220-C001, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-5

Question Source: Bank #

Modified Bank # 862259

New

Question History: System Exam 11 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.8

55.43

Comments:
KA Match: The K/A/ is matched because the examinee must demonstrate knowledge of the operational implications of PZR reference leg leak abnormalities as they apply to Pressurizer Level Control Malfunctions.

High Cog: The question is at the Comprehension/Analysis cognitive level because the examinee must demonstrate understanding of how the Pressurizer Level control system detects level, and then determine how a failure of a reference leg effects actual pressurizer level to correctly answer the question. In addition, the examinee must take into account how an erroneous pressurizer level effects the pressurizer makeup valve which would close and cause actual level to lower.

Plant conditions:

- 100% power.
- Pressurizer Level LT-1 is selected as the controlling channel.

Event:

- A leak develops on the reference leg of LT-1.

Following the event, the trend of indicated (LT-1) Pressurizer level will ___(1)___ and actual Pressurizer level will ___(2)___.

A. (1) rise
(2) lower

B. (1) rise
(2) remain the same

C. (1) lower
(2) rise

D. (1) lower
(2) remain the same

Answer: A

**14** **ID: 1700748** **Points: 1.00**

Which one of the following requires a reactor startup to be aborted?

A. No source range nuclear instrumentation when both intermediate channels reading ~ $5x10^{-11}$ amps.

B. No intermediate range nuclear instrument channels with 2 of 4 power range channels greater than 10% full power.

C. Expected criticality 30% lower than the Estimated Critical Position and on the next pull.

D. Most conservative valid wide range cold leg temperature indication shows RCS temperature less than 530F.

Answer: A

**Answer Explanation**

<<<<Explanation: To answer this question correctly, the examinee must know: (1) In accordancw with Technical Specification 3.5.1.3, the minimum Source Range instrumentation is 1 source range instrument until 1 of 2 intermediate range instruments are greater than $10^{-10}$ amps; (2) If that requirement is not met, the reactor must be shutdown in accordance with technical specification 3.0.1.

| | | |
|---|---|---|
| A. | No source range nuclear instrumentation when both intermediate channels reading ~ $5x10^{-11}$ amps. | CORRECT: Based on T.S. Requirements, Table 3.5-1 |
| B. | No intermediate range nuclear instrument channels with 2 of 4 power range channels greater than 10% full power. | INCORRECT: Plausible because the intermediate range instruments are required UNTIL 2 of 4 power range instruments go over 10% power. The examinee may believe more than two are required to be over 10%, or they are required for longer into the startup. |
| C. | Expected criticality 30% lower than the Estimated Critical Position and on the next pull. | Incorrect: Plausible because there is criteria to abort a reactor startup based on an estimated critical position below the minimum rod withdrawal limit. Incorrect because that criteria is that criticality is expected 30% less than the minimum predicted ECP, not the ECP. |
| D. | Most conservative valid wide range cold leg temperature indication shows RCS temperature less than 530F. | INCORRECT: Plausible because a low RCS temperature is a criteria to abort a reactor startup. The expected temperature that the reactor start is performed is between 530F and 534F. Incorrect because the criteria to abort the startup is less than 525F. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 032 | AK3.01 |
| | Importance Rating | 3.2 | 3.6 |

K/A: Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Startup termination on source-range loss

Proposed Question: RO Question #14

Technical Reference(s): TQ-TM-104-623-C001, Rev 5 Tech Specs Section 3.5.1.3

Proposed References to be provided to applicants during examination: None

Learning Objective: 623-GLO-10

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | |
|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X |
| | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.41 | b.10 |
| | 55.43 | |

Comments:
KA Match: This question matches the K/A because the examinee must know the correlation Loss of SR NI during a startup and the required actions; continue the startup or terminate the startup.

**15** **ID: 1737531** **Points: 1.00**

When required to verify a Radiation Monitoring System Setpoint, which is the PREFERRED METHOD for obtaining a Radiation Monitor's actual setpoints?

A. Contacting the I&C Department to verify the setpoint.

B. Performing a Source Check IAW 1301-4.1, Weekly Surveillance Checks.

C. Reading Alarm setpoints from the meter face by pressing the alarm pushbutton.

D. Refer to the setpoints section of alarm response OP-TM-MAP-C0101, RADIATION LEVEL HI.

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) 1105-8, Radiation Monitoring System a Caution states ***"Reading Alarm setpoints from the meter face by pressing the Alarm Pushbutton does not give an accurate indication of actual setpoint. This feature is not calibrated and should not be relied upon for accurate setpoint information".*** (2) Following that caution a procedure step directs setpoint verification by directing I&C department to refer to applicable sections of 1302-3.1 series procedures that govern RMS Monitors covered by Tech Specs or ODCM and IC-177, RMS Calibration for Non-Tech Monitors.

| | | |
|---|---|---|
| A. | Contacting the I&C Department to verify the setpoint. | CORRECT ANSWER: 1105-8, Radiation Monitoring System Caution and procedure step |
| B. | Performing a Source Check IAW 1301-4.1, Weekly Surveillance Checks. | INCORRECT ANSWER: Plausible because performing a source check will bring in the alarms. However, Tech Spec definition 1.13 defines a source check in the following way: A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source. It does not check the actual setpoints of a radiation monitor. |
| C. | Reading Alarm setpoints from the meter face by pressing the alarm pushbutton. | INCORRECT ANSWER: Plausible because setpoints can be observed when using the pushbutton, however 1105-8 cautions against relying upon it for accurate setpoint information. |
| D. | Refer to the setpoints section of alarm response OP-TM-MAP-C0101, RADIATION LEVEL HI. | INCORRECT ANSWER: Plausible because there is a setpoints section of this alarm response. Incorrect because this section has no setpoints listed and sends the operator to 1101-2.1, RADIATION MONITORING SYSTEM SETPOINTS. |

Examination Outline Cross-reference:

| | | | |
|---|---|---|---|
| Level | RO | | SRO |
| Tier # | 1 | | |
| Group # | 2 | | |
| K/A # | 061 | AA2.03 | |
| Importance Rating | 3.0 | | 3.3 |

K/A: Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Setpoints for alert and high alarms

Proposed Question: RO Question #15

Technical Reference(s): 1105-8, Rev 91

1302-3.1 series procedures

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-9

Question Source: Bank # 887632

Modified Bank #

# EXAMINATION ANSWER KEY
## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

New

Question History: Simulator Exam 9 Last NRC Exam: None

| | | |
|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X |
| | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.41 | b.11 |
| | 55.43 | |

Comments:

KA Match: This matches the KA because the question requires the examinee to determine the method to correctly determine the high setpoint on a radiation monitor. The setpoint on the front of the radiation monitor is not calibrated, and cannot be used to correctly verify the high setpoint.

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

**16** **ID: 1700780** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto..

EVENT:

- MAP C-1-1, RADIATION LEVEL HI alarm illuminates.
- RM-L-1-LO (RCS Letdown Rad Monitor) indicates a HIGH ALARM.
- RM-L-1-HI (RCS Letdown Rad Monitor) indicates an ALERT ALARM.

Based on the given conditions, letdown flow ___(1)___ isolated, because ___(2)___.

A. (1) IS
(2) RM-L-1-LO HIGH alarm closes MU-V-1A and MU-V1B, Letdown Cooler Inlet valves

B. (1) IS
(2) RM-L-1-LO HIGH alarm closes MU-V-2A and MU-V-2B, Letdown Cooler Outlet valves

C. (1) IS NOT
(2) both RM-L-1-LO AND RM-L-1-HI have to be in HIGH Alarm

D. (1) IS NOT
(2) RM-L1-LO HIGH alarm does NOT close MU-V-2A and MU-V-2B

Answer: D

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) RM-L-1-LO and RM-L-1-HI are not the same radiation monitor; (2) RM-L-1-LO has no interlocks; (3) RM-L-1-HI closes MU-V-2A/2B on a HIGH alarm, but not an ALERT alarm; (4) MU-V-1A and MU-V-1B do isolate letdown but there is no interlock between either of the RM-L-1 detectors.

| | | |
|---|---|---|
| A. | (1) IS<br>(2) RM-L-1-LO HIGH closes MU-V-1A and MU-V1B, Letdown Cooler Inlet valves. | INCORRECT: Plausible since the HIGH alarm on some radiation monitors do initiate automatic interlocks, however RM-L-1-LO does NOT have any interlocks associated with its HIGH Alarm. Closing MU-V-1A&1B would isolate letdown flow. However these valves are closed by other interlocks independent of RM-L-1-LO or RM-L-1-HI. |
| B. | (1) IS<br>(2) RM-L-1-LO HIGH alarm closes MU-V-2A and MU-V-2B, Letdown Cooler Outlet valves. | INCORRECT: Plausible since the High Alarm on some radiation monitors do initiate Automatic Interlocks, however RM-L-1-LO does NOT have any interlocks associated with its HIGH Alarm. RM-L-1-HI HIGH alarm closes MU-V-2A and MU-V-2B. |
| C. | (1) IS NOT<br>(2) Both RM-L-1-LO AND RM-L-1-HI have to be in HIGH Alarm. MU-V-2A and MU-V-2B | INCORRECT: Letdown Flow is NOT isolated, BOTH RM-L-1-LO AND RM-L-1-HI do not need to be in HIGH alarm to isolate letdown, just RM-L-1-HI needs to be in HIGH. Plausible because the examinee could believe that a combination of alarms are required to isolate letdown. |
| D. | (1) IS NOT<br>(2) RM-L1-LO "High Alarm" does NOT close MU-V-2A and MU-V-2B. | CORRECT: Letdown Flow IS NOT isolated because MU-V-2A and MU-V-2B remain open since they do NOT close from RM-L-1-LO being in HIGH Alarm. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 1 | | |
| | Group # | 2 | | |
| | K/A # | 076 | AA1.04 | |
| | Importance Rating | 3.2 | | 3.4 |

K/A: High Reactor Coolant Activity: Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity: Failed fuel-monitoring equipment

Proposed Question: RO Question #16

Technical Reference(s): TQ-TM-104-661-C001

1105-8

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-10

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments:

K/A Match: This question matches the KA because the examinee is required to recall the interlocks of RM-L-1-Hi which is the radiation monitor that is used to detected failed fuel. RM-L-1-HI takes a stream off the letdown line, which is RCS water going to the makeup tank.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**17** **ID: 1720243** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- The ICS Selected Feedwater Loop Temperature transmitter slowly fails high.

The Total Feedwater demand signal will ___(1)___ due to ___(2)___.

A. (1) rise
(2) BTU limits

B. (1) rise
(2) Feedwater Temperature Modification

C. (1) lower
(2) BTU limits

D. (1) lower
(2) Feedwater Temperature Modification

Answer: B

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1)There is NO effect on BTU Limits since the transmitter failed high; (2) The operation of the FW temperature circuit compares actual FW temperature (Failed Temperature Transmitter) to a calculated temperature based on Total FW Demand; (3) In this question the circuit would raise FW Demand since the control system thinks Actual FW Temperature is hotter than calculated thus raise FW Flow to match FW FLow heat removal capacity to match Reactor heat output.

| | | |
|---|---|---|
| A. | (1) rise<br>(2) BTU limits | INCORRECT: Plausible since FW Temperature is an input to the BTU Limit Circuit, however, with FW temperature failing high has no effect. If FW temperature had failed low, it would reduce FW Flow. |
| B. | (1) rise<br>(2) Feedwater Temperature Modification | CORRECT: FW temperature Modification is used to correct FW demand for a given load such that there is balance of BTU exchange between the primary and secondary side of the steam generator. So, if FW temp is higher than the calculated expected average FW temp than FW demand signal will rise. |
| C. | (1) lower<br>(2) BTU limits | INCORRECT: Plausible since FW temperature is an input to the BTU limit circuit and would lower FW flow on a low failure. |
| D. | (1) lower<br>(2) Feedwater Temperature Modification | INCORRECT: Plausible if Examinee does NOT understand how the control system process compares actual FW temperature with calculated FW temperature. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | A02 | AK2.1 |
| | Importance Rating | 3.8 | 4.0 |

K/A: Loss of NNI-X/Y: Knowledge of the interrelations between the (Loss of NNI-X) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question: RO Question # 17

Technical Reference(s): OP-TM-104-621-C001, Rev 10 ICS FW Analog & Digital Prints

Proposed References to be provided to applicants during examination: NONE

Learning Objective: 621-GLO-11

Question Source: Bank # 357040

Modified Bank #

New

Question History: System Exam 13 Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of how a Feedwater Temperature failure affects the ICS system for automatic control.

High Cog: This question is High Cog because the examinee has to understand the control system interface, with respect to Main FW control. The examinee must know the control system compares Actual FW Temperature to Calculated Temperature and analyze the effects of a failure.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**18** **ID: 1737211** **Points: 1.00**

To avoid damage due to gas in the Makeup Pumps, __(1)__ must be closed if Makeup Tank level lowers to less than __(2)__ inches with suction from LPI or the BWST.

A. (1) MU-V-18, RCS Makeup Isolation Valve
(2) 18 inches

B. (1) MU-V-12, Makeup Tank Outlet Isolation Valve
(2) 18 inches

C. (1) MU-V-18, RCS Makeup Isolation Valve
(2) 40 inches

D. (1) MU-V-12, Makeup Tank Outlet Isolation Valve
(2) 40 inches

Answer: B

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-211-000 Precautions & Limitations: To avoid pump damage due to gas entrainment in the pump, MU-V-12 must be closed if MU Tank level is less than 18 inches and LPI or BWST is supplying MU Pump Suction; (2) When a Makeup Pump is lined up for suction from LPI or the BWST, the only separation between those sources and the Makeup Tank is MU-V-112, which is a check valve; (3) The Makeup Pump source will be from the source with the highest pressure, which in some instances (based on BWST level or LPI flow) could the the Makeup Tank, which has a low inventory < 18 inches, could cause gas to be admitted to the Makeup Pump.

| | | |
|---|---|---|
| A. | (1) MU-V-18, RCS Makeup Isolation Valve<br>(2) 18 inches | INCORRECT: Plausible if the examinee thinks it is necessary to isolate other Makeup Pump flowpaths to prevent possible Makeup Pump Runout. Part 2 is correct. |
| B. | (1) MU-V-12, Makeup Tank Outlet Isolation Valve<br>(2) 18 inches | CORRECT: See above |
| C. | (1) MU-V-18, RCS Makeup Isolation Valve<br>(2) 40 inches | INCORRECT: Plausible because the minimum level to ensure the Makeup Pump has adequate NPSH when the suction source is the Makeup Tank exclusively is 40". If NPSH were a concern, the examinee could believe lowering the flow by closing MU-V-18 could fix the NPSH issue. |
| D. | (1) MU-V-12, Makeup Tank Outlet Isolation Valve<br>(2) 40 inches | INCORRECT: Plausible since Part 1 is correct and thinks that 40" is the correct setpoint. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 004 | A1.06 |
| | Importance Rating | 3.0 | 3.2 |

K/A: Chemical and Volume Control System: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CVCS controls including: VCT level

Proposed Question: RO Question #18

Technical Reference(s): OP-TM-211-000, Rev 33

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO-14

Question Source: Bank #

Modified Bank #

New X

| | | | |
|---|---|---|---|
| Question History: N/A | | Last NRC Exam: N/A | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | b.5 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must know that the lowering level in the Makeup Tank (VCT) could exceed a design limit for the CVCS system.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**19** **ID: 1737168** **Points: 1.00**

REFERENCE PROVIDED

Plant Conditions:

- Plant Cooldown in Progress
- All 4 Reactor Coolant Pumps are OFF
- DH-P-1A is Operating.
- RCS Temperature = 170F
- RCS Pressure = 250 psig

Curve __(1)__ prevents exceeding LTOP limits. The temperature instrument monitored to prevent exceeding these limits is __(2)__.

A. (1) Curve A
(2) DH6-TI-1 Decay Heat Suction Temperature

B. (1) Curve B
(2) DH6-TI-1 Decay Heat Suction Temperature

C. (1) Curve A
(2) DH2-TI-1 Decay Heat Cooler Outlet Temperature

D. (1) Curve B
(2) DH2-TI-1 Decay Heat Cooler Outlet Temperature

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) The 1102-11 Limits and Precautions directs reactor coolant temperature to be maintained within limits specified in Enclosure 4, Figure 1 and 1A one of which is the LTOP and SURGE Line Limit (CURVE A); (2) The operators must ensure the pressure and temperature remain to the below and to the right of CURVE A; (3) The examinee must know the correct instruments to monitor to ensure the cooldown rate and LTOP limits are not violated, which for these plant conditions (RCPs secured, while on DHR) is DH2-TI1 or 2; (4) CURVE B is the Spray Restriction line.

| | | |
|---|---|---|
| A. | (1) Curve A<br>(2) DH6-TI-1 Decay Heat Suction Temperature | INCORRECT: Plausible since the operator may assume DH Suction Temperature used during a cooldown. Incorrect because after RCPs are secured DH Cooler Outlet Temp is used. Curve A is correct. |
| B. | (1) Curve B<br>(2) DH6-TI-1 Decay Heat Suction Temperature | INCORRECT: Plausible since the operator may assume DH Suction Temperature used during a cooldown. Incorrect because after RCPs are secured DH Cooler Outlet Temp is used. Curve B is incorrect. |
| C. | (1) Curve A<br>(2) DH2-TI-1 Decay Heat Cooler Outlet Temperature | CORRECT: See above |
| D. | (1) Curve B<br>(2) DH2-TI-1 Decay Heat Cooler Outlet Temperature | INCORRECT: Plausible since the temperature indication is correct but CURVE B is the Spray Restriction Curve. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 2 | |
| Group # | 1 | |
| K/A # | 005 | K5.01 |
| Importance Rating | 2.6 | 2.9 |

K/A: Residual Heat Removal System: Knowledge of the operational implications of the following concepts as they apply the RHRS: Nil ductility transition temperature (brittle fracture)

Proposed Question: RO Question #19

Technical Reference(s): TQ-TM-104-212-C001, Rev 17 Tech Spec 3.1.2

1102-11, Rev 153B

Proposed References to be provided to applicants during examination: 1102-11 Fig 1A

Learning Objective: 212-GLO-10

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | |
|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X |
| | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.41 | |
| | 55.43 | b.5 |

Comments:

KA Match: This question matches the KA because the reference curves in 1102-11 are developed in ensure that NDT and cyclic stress limits are not violated. The examinee must know which instrument is used to ensure the LTOP curve A is NOT violated.

# EXAMINATION ANSWER KEY
16-01 SENIOR REACTOR OPERATOR NRC EXAM

**20** **ID: 1700999** **Points: 1.00**

Plant conditions:

- Plant Cooldown is in progress in accordance with 1102-11.
- RCS pressure is 60 psig and steady prior to venting.
- OP-TM-220-552, VENTING THE PRESSURIZER TO THE RCDT (WDL-T-3), has been initiated to lower pressurizer pressure.

To prevent large pressure transients during the Pressurizer venting, the operator must ________

A. open RC-V-18, Manual Pzr Vent to RCDT.

B. throttle open RC-V-28, Pressurizer Vent Valve, for 1 second.

C. throttle open RC-V-1, Spray Valve, to help lower Pressurizer pressure.

D. ensure RCDT cooling pump (WDL-P-8) is in AUTO with both ICCW Pumps running.

Answer: B

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) During and plant cooldown, to lower RCS pressure in the latter stages, the pressurizer is vented to the RCDT; (2) To limit pressure and/or level transients, RC-V-28 is throttled open in small amounts (1 second per procedure); (3) At 60 psig in the cooldown, no RCPs are running so spray from RC-V-1 is not available.

| | | |
|---|---|---|
| A. | open RC-V-18, Manual Pzr Vent to RCDT. | INCORRECT: Plausible since this is a vent valve to the RCDT; however it is not allowed to be open when the RCS is > 45 psig to prevent overpressurizing the RCDT. |
| B. | throttle open RC-V-28, Pressurizer Vent Valve, for 1 second. | CORRECT: A CAUTION in OP-TM-220-552 says to use small taps on RC-V-28 open pushbutton to avoid large pressure or level transients. The procedure step is to throttle open for 1 second. |
| C. | throttle open RC-V-1, Spray Valve, to help lower Pressurizer pressure. | INCORRECT: Plausible since RC-V-1 is the normal spray valve; however the last RCP was shut down prior to the venting operation being initiated. |
| D. | ensure RCDT cooling pump (WDL-P-8) is in AUTO with both ICCW Pumps running. | INCORRECT: Plausible since two ICCW Pumps will be run if the RCS temperature was >400°F; however WDL-P-8 will be started and run continuously during the venting operation and the RCS temperature cannot be > 400°F based on the given condition of 60 psig in the RCS. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 007 | 2.1.23 |
| | Importance Rating | 4.3 | 4.4 |

K/A: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: RO Question # 20

Technical Reference(s): OP-TM-220-000, Rev 23
OP-TM-220-552, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-10

Question Source: Bank # 8672881
Modified Bank #
New

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

| | | |
|---|---|---|
| | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.41 | b.10 |
| | 55.43 | |

Comments:
K/A Match: This question matches the KA because the examinee will have do demonstrate knowledge regarding the ability to mitigate large pressure transients during and RCS cooldown. The examinee will have to have knowledge of the procedure to vent the Pressurizer to the RCDT.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**21** **ID: 1701024** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- EFW Area Cooling Fan, AH-E-24A, is operating.
- EFW Area Cooling Fan, AH-E-24B, is in standby.
- Emergency Feedwater Pump, EF-P-2A is running for an IST.

EVENT:

- Nuclear Service Closed Cooling Water Inlet Temperature Controller to AH-E-24A Coolers, TC-857, Fails LOW.

As a result of this controller failure, identify the resulting effect(s):

AH-E-24A trips due to high __(1)__ temperature, and AH-E-24B __(2)__.

A. (1) outlet
(2) automatically starts

B. (1) outlet
(2) can be started manually

C. (1) motor
(2) automatically starts

D. (1) motor
(2) can be started manually

Answer: B

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) AH-E-24A/24B have NO autostart features; (2) Fan outlet temperature is monitored by a temperature Switch TS-766A/766B; (3) The fan Motor does NOT have a Temperature Switch; (4) Since the NSCCW Temperature Controller failed LOW, the affect would reduce NSCCW Flow to the fan coolers resulting in fan outlet temperature rising (sensed by TS-766A); (5) At 125F the operating Fan would trip; (6) Alarm response HVA-5-8 would direct starting of the standby fan as long as there is no fire present (the stem of the question gives no indication of a fire).

| | | |
|---|---|---|
| A. | (1) outlet<br>(2) automatically starts | INCORRECT: With the NSCCW Temperature Controller failed low, NSCCW inlet valve would close which would reduce cooling flow to the Fan Cooling Coils. This would result in fan discharge temperature rising. At 125F the fan would trip. AH-E-24B does not have an autostart feature. This is plausible because many components have autostart features on temperature and/or a trip. |
| B. | (1) outlet<br>(2) can be started manually | CORRECT: With the NSCCW temperature controller failed low, NSCCW inlet valve would close which reduces cooling flow to the fan cooling coils resulting in fan discharge temperature rising. At 125F the fan would trip. AH-E-24B does not have an autostart feature and could be manually started. |
| C. | (1) motor<br>(2) automatically starts | INCORRECT: Plausible since an operator may believe that the fan would trip on high motor temperature as opposed to fan discharge temperature. Since AH-E-24B is in standby, it would have to be manually started since there is no autostart feature on these fans. |
| D. | (1) motor<br>(2) can be started manually | INCORRECT: Plausible since an operator may believe that the fan would trip on high motor temperature as opposed to fan discharge temperature. Since AH-E-24B is in standby, it could be manually started since there is no autostart feature on these fans. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 008 | A4.09 |
| | Importance Rating | 3.0 | 2.9 |

K/A: Component Cooling Water System: Ability to manually operate and/or monitor in the control room: CCW temperature control valve

Proposed Question: RO Question # 21

Technical Reference(s): HVA-5-8 Rev 9

TQ-TM-104-821-C001, Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: 821-GLO-010

Question Source: Bank #

| | | | | |
|---|---|---|---|---|
| | Modified Bank # | | | |
| | New | X | | |
| Question History: | N/A | | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | | X | |
| 10 CFR Part 55 Content: | 55.41 | b.7 | | |
| | 55.43 | | | |

Comments:

KA Match: This question matches the KA because the examinee will have to know that the operator has to manually operate AH-E-24B from the control room if the temperature control valve for the CCW to the AH-E-24A fan were to fail.

High Cog: This question is high cog because the examinee has to identify the effect of the temperature controller failing low and the response of the associated control valve. This would result in the control valve closing, reducing cooling flow. The fan outlet temperature would rise, which is the signal monitored to for trip the operating fan.

**22** **ID: 1701070** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Pressurizer heater pressure control SETPOINT (RC3-PIC) signal fails to zero (0) psig.

What action(s) are required?

A. Raise pressurizer level to maintain pressure until heaters are restored.

B. Place the Pressurizer Spray Valve, RC-V-1 in Manual and CLOSE RC-V-1.

C. Verify all pressurizer heaters are deenergized OR manually deenergize heaters.

D. Adjust Heater Banks 1, 2 & 3 using RC3PIC, Pressurizer Pressure control in hand.

Answer: D

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<<<Explanation: To answer this question correctly, the examinee must know: (1) That the SETPOINT signal only affects PZR Heater Banks 1, 2, & 3 while the controller is in auto (2) the controller failing to 0 psig means heaters will not start at the desired setpoint (3) the crew can place the Pzr Controller in HAND per OS-24 and operate per OP-TM-220-503, "Manual Control of the Pzr" at that point the pressure lowering transient is concluded. Even though OP-TM-220-503 does NOT address instrument failures, control of RCS pressure is ensured more quickly and smoothly by entering and performing this procedure. (4) The mental process involved is that RCS pressure will lower if heaters are not on to account for losses.

| | | |
|---|---|---|
| A. | Raise pressurizer level to maintain pressure until heaters are restored. | INCORRECT - Plausible because raising pressurizer level will squeeze the pressurizer steam space, raising (or maintaining) RCS pressure. Incorrect because the guidance for an action is in OP-TM-AOP-043, of which, the entry criteria is NOT met. |
| B. | Place the Pressurizer Spray Valve, RC-V-1 in Manual and CLOSE RC-V-1. | INCORRECT - Plausible if the analysis of the failure causes the examinee to believe that the spray valve could erroneously open. This could happen on other pressure instrument/controller failures. |
| C. | Verify all pressurizer heaters are deenergized OR manually deenergize heaters. | INCORRECT - Plausible if the analysis of the failure causes the examinee to believe that the setpoint failing low energizes the heaters. This could happen on other pressure instrument/controller failures. |
| D. | Adjust Heater Banks 1, 2 & 3 using RC3PIC, Pressurizer Pressure control in hand. | CORRECT - See above. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 1 | |
| Group # | 1 | |
| K/A # | 027 | AA1.01 |
| Importance Rating | 4.0 | 3.9 |

K/A: Pressurizer Pressure Control System Malfunction: Ability to operate and / or monitor for the following as they apply to the Pressurizer Pressure Control Malfunctions: PZR heaters, sprays, and PORVs

Technical Reference(s): OP-TM-220-503, Rev 4

Proposed Question RO Question #22

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-11

Question Source: Bank # 897217

Modified Bank #

New

# EXAMINATION ANSWER KEY
16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question History: | | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | 7 | |
| | 55.43 | | |

Comments:

KA MATCH: This is KA match because the operate will have to understand that heaters must be operated in manual with this controller failure.

HIGH COG: This is High Cog because they will have to realize the heater setpoint failing to zero will only effect the pressurizer heaters and then that manual operation is required to control the plant.

**23** **ID: 1701075** **Points: 1.00**

Initial plant conditions:
- Reactor trip due to LOCA.
- Containment Building pressure 5 psig, steady.
- Manual ESAS actuation signals were NOT initiated.
- Automatic ES Actuation status:

| ES Actuation | Train A | Train B |
|---|---|---|
| 1600# | Actuated (NOT Bypassed) | Bypassed |
| 500# | Not Actuated | Not Actuated |
| 4# | Defeated | Actuated (NOT Defeated) |

EVENT:
- Containment Building pressure rises rapidly to 35 psig.

Based on these conditions, identify the ONE statement below that describes the response of the Reactor Building Spray System

A. Only BS-P-1A starts.

B. Only BS-P-1B starts.

C. Both BS-P-1A and BS-P-1B start.

D. Neither BS-P-1A nor BS-P-1B starts.

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) RB Pressure of 30 psi is ONE input to the Start Signal for a building spray pump; (2) The other requirement is a Block 4 permissive signal from any or the following ESAS Actuation signals: 1600 psig OR 500 psig OR 4 psig; (3) While the 4 psig signal is the only signal that opens the BS valves the other signals (1600 psig & 500 psig) input to the Block 4 permissive signal; (4) The BS Pump start signal does not require the BS valves to be open for the pump to start.

| | | |
|---|---|---|
| A. | Only BS-P-1A starts. | INCORRECT: Distracter is plausible because it acknowledges BS-P-1A starts due to ES Block 4 permit from 1600 psi. This distracter is based on the misconception that BS pumps require a Block 4 start permit (from 1600/500# ES actuation) in order to automatically start at 30 psig. |
| B. | Only BS-P-1B starts. | INCORRECT: Distracter is plausible because it acknowledges that BS-P-1B will start. This distracter is based upon the common misconception that BS pumps require a Block 4 start permit from 4# ES actuation (opens BS valves) in order to start at 30 psig. |
| C. | Both BS-P-1A and BS-P-1B start. | CORRECT: Both BS Pumps start since both the 1600 psi (Train A) and 4 psi (Train B) signals provide Block 4 permissive signals to enable the BS Pumps to start at 30 psi. |
| D. | Neither BS-P-1A and BS-P-1B start. | INCORRECT: Plausible since Part of Train A (4 psi) is in Defeat and part of Train B (1600 psi) is Bypassed preventing Block 4 BS Pump Permissive Signal. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 013 | K1.01 |
| | Importance Rating | 4.2 | 4.4 |

K/A: Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: Initiation signals for ESF circuit logic

Proposed Question: RO Question #23

Technical Reference(s): TQ-TM-104-642-C001, Rev 7 ESAS Electrical Prints

Proposed References to be provided to applicants during examination: None

Learning Objective: 642-GLO-10

Question Source: Bank # 371299

Modified Bank #

New

| | | | |
|---|---|---|---|
| Question History: N/A | | Last NRC Exam: N/A | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments: (KA Match, why high cog, why SRO only)
KA MATCH: This question matches the KA because the examinee must have knowledge of the initiation signals which start the Reactor Building Spray pumps, which is an ES system.

HIGH COG: This question is High Cog because the examinee must analyze the reactor building conditions and the actuated/defeated ES signals to determine which (if any) reactor building spray pumps start.

**24** **ID: 1701092** **Points: 1.00**

REFERENCE PROVIDED

Plant Conditions:

- 100% power with ICS in full auto.
- RB Cooling Fans AH-E-1A/B/C are all operating in FAST SPEED.

EVENT:

- THERMAL OVERLOAD (49F) actuates.
- Thermal overload condition has NOT been reset at AH-E-1A breaker.

SUBSEQUENT EVENT:

- Main Steam System leak inside the Containment Building.
- RB Temperature Elevation > 320 is at 132F and slowly rising.
- NN-2-7 RB AIR TEMP HI is in Alarm.
- RB Pressure 2.3 psig.

Based on these conditions, AH-E-1A is __(1)__. In accordance with OP-TM-534-901, RB EMERGENCY COOLING OPERATIONS, AH-E-1A must be started in /shifted to __(2)__ speed.

A. (1) tripped
(2) slow

B. (1) running
(2) slow

C. (1) tripped
(2) fast

D. (1) running
(2) fast

Answer: A

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

Explanation: To answer this question correctly, the examinee must know: (1) Separate thermal overload contacts exist for slow and fast speed operation (see reference 208-561, 49F/S contacts in blocks B6 and B5); (2) Operation at slow speed would not be affected by the fast speed thermal overload contact tripping; (3) Since RB Pressure is > 2 psig, the procedure requires the fan be manually started in slow speed; (4) In this case, the operating crew must start RB Emergency cooling, then when pressure is greater than 2.0 psig, the fans are shifted/started in slow speed.

| | | |
|---|---|---|
| A. | (1) tripped<br>(2) slow | CORRECT: Refer to 208-561. Two independent thermal overload contacts, one for slow speed operation and one for fast speed operation. The fan will trip and fast speed. When the crew puts RB Emergency cooling in service, the operator will start the fan in slow speed when RB pressure becomes > 2 psig. |
| B. | (1) running<br>(2) slow | INCORRECT: Plausible because the examinee must identify in the drawing that the 49F thermal overload does not affect the slow speed motor. The examinee could believe that the 49F is for the slow speed motor. |
| C. | (1) tripped<br>(2) fast | INCORRECT: Plausible because the examinee could believe the fan is started in fast speed when RB pressure becomes > 2 psig. In addition, the examinee must use the drawing to determine the effect of the 49F thermal overload actuation. |
| D. | (1) running<br>(2) fast | INCORRECT: Plausible because the examinee must determine the effect of the 49F thermal overload actuation on the operation of the fan. In addition, the examinee could believe that the fan must be started in fast speed when RB pressure becomes > 2 psig. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 2 | |
| Group # | 1 | |
| K/A # | 022 | A2.03 |
| Importance Rating | 2.6 | 3.0 |

K/A: Containment Cooling System: Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Fan motor thermal overload/high-speed operation

Proposed Question: RO Question #24

Technical Reference(s): Electrical Print 208-561, ~~208-461~~ 208-416 JO 6/23/17 MAP NN-2-7 alarm response, Rev 2

OP-TM-534-901 Rev 14

Proposed References to be provided to applicants during examination: Electrical Prints 208-561 208-461

Learning Objective: 824-GLO-10

Question Source: Bank #

Modified Bank # 371313

New

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.5

55.43

Comments:
KA Match: This question matches the KA because the examinee must be able to predict the impact of an actuated thermal overload on the normal operation of the RB Emergency cooling fan and what the speed the fan must be run in accordance with OP-TM-534-901.

HIGH Cog: This question is high cog because the examinee must analyze the given station drawing to predict the impact of the actuated thermal overload and identify the procedure requirement for the given plant conditions.

Plant conditions:
- Reactor is operating at 100% power, with ICS in full automatic.
- RB Cooling Fans AH-E-1A/B/C are all operating in FAST SPEED.
- Main Steam System leak inside the Containment Building.

Event:
- AH-E-1A trips due to THERMAL OVERLOAD.
- Thermal overload condition has NOT been reset at AH-E-1A breaker.

Based on these conditions, identify the ONE selection below that describes impact on AH-E-1A operations:
(1) Ability to manually start AH-E-1A in SLOW speed;
(2) Automatic ES operation.
(3) Reason for impact.

A. (1) Can be manually started in slow speed.
(2) Will automatically start if ES actuates.
(3) Thermal overload protection only applies to FAST speed operation.

B. (1) Can be manually started in slow speed.
(2) Will automatically start if ES actuates.
(3) SEPARATE thermal overload protection is provided for the high and slow speed motor contactors.

C. (1) CANNOT be manually started in slow speed.
(2) Will automatically start if ES actuates.
(3) COMMON thermal overload protection (for fast AND slow speeds) is DEFEATED by ES actuation.

D. (1) CANNOT be manually started in slow speed.
(2) Will NOT automatically start if ES actuates.
(3) COMMON thermal overload protection (for fast AND slow speeds) is NOT defeated by ES actuation.

Answer: B

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**25** **ID: 1718554** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- OP-TM-621-251 SASS Logic Test in Progress.
  - Test 'A' on NI-5 / NI-6 Rx Pwr was successful, however the SASS channel FAILED to reset.
- NI-5 is selected for control.

EVENT:

- NI-5 upper chamber power supply fails to zero volts.

The crew __(1)__, because __(2)__.

A. (1) must immediately enter OP-TM-AOP-070, Primary to Secondary Heat Transfer Upset
(2) reactor power would be lowering

B. (1) must immediately enter OP-TM-AOP-070, Primary to Secondary Heat Transfer Upset
(2) reactor power would be rising

C. (1) must immediately enter OP-TM-EOP-001, Reactor Trip
(2) the NI-5 failure would cause a reactor trip

D. (1) would have NO required actions
(2) the SASS would automatically transfer to NI-6 for reactor control

Answer: B

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) When the SASS channel fails to reset, that the SASS channel would not swap to a good instrument on a failure of the controlling instrument; (2) When the NI-5 upper chamber power supply fails to zero volts, the NI-5 indicated power drops to 50%; (3) Because the SASS is failed, when NI-5 fails low, the an ICS transient would occur that pulls control rods to maintain reactor power; (4) When control rods pull, reactor power would rise; (5) NI-5 could reach an RPS trip setpoint, but this would be the only immediate reactor trip signal; (6) To mitigate this plant transient, the crew would enter OP-TM-AOP-070 because a plant transient would be occurring requiring manual operation, and no valid ICS runback is present.

| | | |
|---|---|---|
| A. | (1) must immediately enter OP-TM-AOP-070, Primary to Secondary Heat Transfer Upset (2) reactor power would be lowering | INCORRECT: The entry criteria for OP-TM-AOP-070 will be met, but reactor power would be rising. This is plausible if the examinee believes NI-5 failing low would result in plant runback. |
| B. | (1) must immediately enter OP-TM-AOP-070, Primary to Secondary Heat Transfer Upset (2) reactor power would be rising | CORRECT: See above. |
| C. | (1)must immediately enter OP-TM-EOP-001, Reactor Trip (2) the NI-5 failure would cause a reactor trip | INCORRECT: Plausible if examinee believes if the examinee believes the NI-5 failure input to RPS would initiate a reactor trip which would require entry into OP-TM-EOP-001. If OP-TM-AOP-070 was not entered, a reactor trip could occur, but this failure in itself would not cause a reactor trip. |
| D. | (1) would have NO required actions (2) the SASS would automaticaly transfer to NI-6 for reactor control | INCORRECT: Plausible because this is how the system should work if the NI-5 upper chamber power supply failed low. Incorrect because the SASS channel failed to reset in the stem of the question. |

Examination Outline Cross-reference:

| Level | RO | | SRO |
|---|---|---|---|
| Tier # | 2 | | |
| Group # | 2 | | |
| K/A # | 015 | A2.01 | |
| Importance Rating | 3.5 | | 3.9 |

K/A: Nuclear Instrumentation System: Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power supply loss or erratic operation

Proposed Question: RO Question # 25

Technical Reference(s): TQ-TM-104-621-C001 OP-TM-AOP-070, Rev 5

OP-TM-621-251 Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: 621-GLO-10

| | | | | |
|---|---|---|---|---|
| Question Source: | Bank # | | | |
| | Modified Bank # | | | |
| | New | X | | |
| Question History: | N/A | | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | | | |
| | 55.43 | | | |

Comments:

KA Match: This question matches the KA because the examinee must know the effects of a failing power supply on plant operations, and choose the procedure to mitigate the failure.

High Cog: This question is high cog because the examinee must analyze the effect of the failure and determine plant response and determine that the response will meet the threshold for the entry criteria of OP-TM-AOP-070.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**26** **ID: 1720398** **Points: 1.00**

Which one of the following describes the operation of the Main Feedwater Pump Monitoring circuit that provides Emergency Feedwater with an input from the Heat Sink Protection System?

EACH PUMP uses:

A. Two pressure bistables to sense Main Feedwater Pump discharge pressure.

B. Three pressure bistables to sense Main Feedwater Pump discharge pressure.

C. Two pressure bistables to sense hydraulic oil pressure at the Main Feedwater Pump turbine stop valves.

D. Three pressure bistables to sense hydraulic oil pressure at the Main Feedwater Pump turbine stop valves.

Answer: C

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) Each HSPS train monitors one pressure bistable at each MFW pump; (2) Both pressure bistables on each MFW pump lower below the pressure setpoint for to get a signal to each HSPS train; (3) When an HSPS train sees both MFW pumps tripped, it will send an actuation signal to that trains associated EFW components.

| | | |
|---|---|---|
| A. | Two pressure bistables to sense Main Feedwater Pump discharge pressure. | INCORRECT: This is plausible because the presence of Feed Pump discharge pressure would be indicative of the operating status of the Main Feed Pump. The operator may incorrectly believe that the parameter sensed to determine that the FW Pump is operating or not is the discharge pressure. |
| B. | Three pressure bistables to sense Main Feedwater Pump discharge pressure. | INCORRECT: This is plausible because the presence of Feed Pump discharge pressure would be indicative of the operating status of the Main Feed Pump. The operator may incorrectly believe that the parameter sensed to determine that the FW Pump is operating or not is the discharge pressure. |
| C. | Two pressure bistables to sense hydraulic oil pressure at the Main Feedwater Pump turbine stop valves. | CORRECT: See above. |
| D. | Three pressure bistables to sense hydraulic oil pressure at the Main Feedwater Pump turbine stop valves. | INCORRECT: This is plausible because there are nine bistables per pump, three of them are associated with the Main Turbine Trip Circuit. The operator may incorrectly believe that the three bistables are associated with the HSPS as well. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 1 | | |
| | K/A # | 061 | K1.02 | |
| | Importance Rating | 3.4 | | 3.7 |

K/A: Auxiliary / Emergency Feedwater System: Knowledge of the physical connections and/or cause-effect relationship between the AFW and following systems: MFW System

Proposed Question: Question 26

Technical Reference(s): TQ-TM-104-644-C001

Proposed References to be provided to applicants during examination: None

Learning Objective: 644-GLO-3

Question Source: Bank # 909174

Modified Bank #

New

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

Question History: Last NRC Exam: 10-02

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments: (KA Match, why high cog, why SRO only)

KA Match: This question matches the KA because the operator must demonstrate knowledge of the physical connections and/or cause-effect relationships between the EFW and the MFW System. This is accomplished by requiring that the operator identify the number of bistables used in the MFW Pump monitoring circuit, and the parameter that is sensed to determine that the MFW Pumps are tripped to automatically start the EFW Pumps.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**27** **ID: 1737222** **Points: 1.00**

Plant conditions:

- Reactor critical at $10^{-8}$ amps.
- Crew is performing a middle of cycle startup from a 3 week forced outage.
- ICS in HAND, EXCEPT for the Main & Startup Feedwater Valves and Turbine Bypass Valves.
- Power escalation is on hold; will recommence in one hour.

EVENT:

- MS-V-3D, Turbine Bypass Valve fails OPEN

Assume NO Operator actions.

Which statement below describes the plant response to this failure?

A. RCS pressure lowers and the Reactor trips on low pressure.

B. Reactor power increases to the 'Point of Adding Heat' and stabilizes at ~1%.

C. Reactor power increases above the 'Point of Adding Heat' and then stabilizes at ~3%.

D. Reactor power increases to the 'Point of Adding Heat' and then lowers to $10^{-8}$ Amps in the Intermediate Range.

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) With the plant stable at 10E-8 AMPS on the IR and when TBV fails open, this would induce a positive reactivity feedback due to the Moderator Temperature Coefficient; (2) Reactor Power would rise to the Point of Adding Heat and continue to rises as Tave lowers; (3) Eventually the power rise will stop and begin to lower as T-ave begins to rise; (3) Reactor power will stabilize at a lower power (~3%), based on the capacity of the TBV.

| | | |
|---|---|---|
| A. | RCS pressure lowers and the Reactor trips on low pressure. | INCORRECT: Plausible since RCS pressure will lower due to the overcooling, however, RCS pressure does NOT reach the the low pressure trip setpoint with a single TBV failing open. |
| B. | Reactor Power increases to the 'Point of Adding Heat' and stabilizes at ~1% Reactor Power. | INCORRECT: Plausible since reactor power does rise to the Point of Adding Heat(~1%), however it does NOT stabilize nor return to 1%. |
| C. | Reactor power increases above the 'Point of Adding Heat' and then stabilizes at ~3%. | CORRECT: See above |
| D. | Reactor power increases to the 'Point of Adding Heat' and then lowers to $10^{-8}$ Amps in the Intermediate Range. | INCORRECT: Plausible since reactor power does rise to the Point of Adding Heat(~1%) and above, and the examinee may think that eventually Reactor Power will then start to lower and stabilize where the transient started. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 1 | | |
| | K/A # | 039 | K5.08 | |
| | Importance Rating | 3.6 | | 3.6 |

K/A: Main and Reheat Steam System: Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity

Proposed Question: RO Question # 27

Technical Reference(s): GFES: Reactor Operational Physics

TQ-TM-104-411-C001, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-10

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.5 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee will have to know that when a turbine bypass valve opens up, when the reactor is below the POAH, that the RCS temperature will drop. The RCS temperature drop, will result in positivity reactivity addition, indicated by a +SUR on the IR Instrument. Reactor Power will rise above the Point of Adding Heat and eventually stabilize at ~3% Power on the NI's. The Three Mile Island systems knowledge being tested is the capacity of the Turbine Bypass Valves.

High Cog: This question is high cog because the examinee has to relate the component failure to the effect on RCS temperature and then the effect the change in RCS temperature has on reactivity feedback.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**28** **ID: 1717325** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- EF-P-2B, Motor Driven Emergency Feedwater Pump, is out of service for maintenance.

EVENT:

- All four Reactor Coolant Pumps trip.
- EF-P-1, Steam Driven Emergency Feedwater Pump, tripped on overspeed.
- 1D 4160V ES bus tripped on overcurrent.
- Subcooling margin is 35F and slowly lowering.
- Incore temperatures are rising.
- 'A' OTSG Level is at 36% Operating Range
- 'B' OTSG Level is at 30% Operating Range

Based on these plant conditions, the operator is required to feed the OTSG's with __(1)__ at a rate of __(2)__ to promote __(3)__.

A. (1) Main Feedwater
(2) >1 Mlbm/hr
(3) Natural Circulation

B. (1) Main Feedwater
(2) >1 Mlbm/hr
(3) Boiler Condenser Cooling

C. (1) Emergency Feedwater
(2) > 215 GPM /OTSGs
(3) Boiler Condenser Cooling

D. (1) Emergency Feedwater
(2) < 515 GPM
(3) Natural Circulation

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) When all four reactor coolant pumps trip, the reactor trips and enters the examinee into the EOP network; (2) Since EF-P-2B is out of service, EF-P-1 trips, and the 1D 4160V bus trips on overcurrent, there are no Emergency Feedwater pumps running, but the Main Feedwater Pumps are still running; (3) After the trip OP-TM-EOP-001 directs the ARO to feed the OTSG's in accordance with OP-TM-EOP-010, Rule 4; (4) From Rule 4, since subcooling margin is >25F, there is NO requirement to raise level to 75% to 85%; (5) But since there are no reactor coolant pumps operating and and no emergency feedwater, feed with MFW at >1Mlb/hr.

| | | |
|---|---|---|
| A. | (1) Main Feedwater (2) >1Mlbm/hr (3) Natural Circulation | CORRECT: See above |
| B. | (1) Main Feedwater (2) >1Mlbm/hr (3) Boiler Condenser Cooling | INCORRECT: Plausible because the examinee could believe that since no reactor coolant pumps are running, that feeding the OTSGs at > 1Mlbm/hr is setting the plant up for boiler condenser cooling. Incorrect because the OTSGs are being setup to promote natural circulation. |
| C. | (1) Emergency Feedwater (2) >215 GPM /OTSGs (3) Boiler Condenser Cooling. | INCORRECT: Plausible because the examinee could believe that Emergency Feedwater is not lost. If Emergency Feedwater was available, and subcooling margin were less than 25F the feedrate would be > 215 GPM/OTSG. |
| D. | (1) Emergency Feedwater (2) < 515 GPM to both OTSGs (3) Natural Circulation | INCORRECT: Plausible because the examinee could believe that Emergency Feedwater is not lost. If Emergency Feedwater was available, and only EF-P-2A or EF-P-2B were running, there would be a limit of < 515 GPM. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 061 | K3.01 |
| | Importance Rating | 4.4 | 4.6 |

K/A: Auxiliary/Emergency Feedwater System: Knowledge of the effect that a loss or malfunction of the AFW will have on the following: RCS

Proposed Question: RO Question #28

Technical Reference(s): OP-TM-EOP-010, Rule 4, Rev 19

OP-TM-EOP-0101, Rule 4 basis, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: 424-GLO-11

Question Source: Bank #

Modified Bank #

| | | | |
|---|---|---|---|
| | New | X | |
| Question History: | N/A | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments: (KA Match, why high cog, why SRO only)
KA MATCH: This question matches the KA because the AFW (EFW for TMI) malfunction results in the operator feeding the OTSGs at a different rate than if all of the EFW pumps were running.

High Cog: This question is high cog because the examinee will have to analyze the event in the stem to realize that the reactor is tripped, and that there are no EFW pumps running.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**29** **ID: 1717742** **Points: 1.00**

In-accordance-with 1107-2C Vital DC Electrical System, which of the following local indications will tell an operator that a battery charger is in the EQUALIZE mode?

A. The AC Pilot Light is lit.

B. The current flow is at least 50 amps.

C. The output voltage is approximately 135 volts.

D. Both the AC & DC input and output breakers are closed.

Answer: C

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1107-2C, the operator observes each charger increases voltage to 135 ± 2 volts.

| | | |
|---|---|---|
| A. | The AC Pilot Light is lit | INCORRECT: Plausible if the examinee is NOT familiar with Battery Charger indication. The AC Pilot light is normally LIT. |
| B. | The current flow is at least 50 amps | INCORRECT: Plausible since current flow indication is available on the Battery Charger. Also, if inverters are on the DC power supply, current could be > 50 amps even in float mode. |
| C. | The output voltage is approximately 135 volts. | CORRECT: 1107-2C: Observe that the voltmeter on the front of each charger increases to 135 ± 2 volts. |
| D. | Both the AC & DC input and output breakers are closed. | INCORRECT: Plausible because both of these breakers are closed at all times of Battery Charger operation. Incorrect because these breakers being closed is not an exclusive indication that a battery charger is in the EQUALIZE mode. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 1 | | |
| | K/A # | 063 | A3.01 | |
| | Importance Rating | 2.7 | | 3.1 |

K/A: D.C. Electrical Distribution: Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights

Proposed Question: RO Question #29

Technical Reference(s): 1107-2C, Rev 12

Proposed References to be provided to applicants during examination: None

Learning Objective: 734-GLO-2

Question Source: Bank # 356975

Modified Bank #

New

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

| |
|---|
| 55.43 |
| Comments: (KA Match, why high cog, why SRO only)<br>KA Match: This question matches the KA because the examinee must know which of the multiple indications listed at possible answer is correct for an equalizing charge (part of the DC distribution system). |

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**30** **ID: 1717423** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- 1M DC Bus is Powered from 1A DC Distribution Panel.

EVENT:

- Steam leak in the Reactor Building.
- The Reactor is tripped.
- RB Pressure increases to 4.2 psig.

10 Minutes Later:

- Loss of Offsite Power

Based on the above plant conditions:

__(1)__transfer capability of 1M DC Bus to its alternate power supply is blocked.

This transfer capability is blocked due to the __(2)__ signal.

A. (1) Only the auto, (2) undervoltage

B. (1) Only the auto, (2) Engineered Safeguards

C. (1) Both the auto and manual, (2) under voltage

D. (1) Both the auto and manual, (2) Engineered Safeguards

Answer: D

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) The design is for the 1M DC 480V Bus transfer to its alternate power supply when its normal power supply is lost on an undervoltage; (2) Any 1600 psig, 500 psig, or 4 psig ES signal blocks the auto and manual transfer if power were lost to 1M DC.

| | | |
|---|---|---|
| A. | (1) Only the auto,<br>(2) undervoltage | INCORRECT: Plausible if the examinee has the misconception that the Loss of Offsite Power prevents the auto transfer of 1M DC. Since AC power is lost, the examinee could believe that 1M DC transfer is prohibited for reliability. 1M DC is still able to transfer to the alternate power supply when plant conditions/procedures allow. |
| B. | (1) Only the auto,<br>(2) Engineered Safeguards | INCORRECT: Plausible if the examinee has the misconception that the only the auto transfer is blocked with an ES Actuation signal. The examinee could believe that the operators could still transfer 1M manually. Incorrect because the auto and manual transfers are blocked on an ES signal. |
| C. | (1) Both the auto and manual,<br>(2) under voltage | INCORRECT: Plausible if the examinee has the misconception that the Loss of Offsite Power prevents auto transfer of 1M DC. Since AC power is lost, the examinee could believe that 1M DC transfer is prohibited for reliability. |
| D. | (1) Both the auto and manual,<br>(2) Engineered Safeguards | CORRECT: See above |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 2 | |
| Group # | 1 | |
| K/A # | 063 | K4.02 |
| Importance Rating | 2.9 | 3.2 |

K/A: DC Electrical Distribution: Knowledge of DC electrical system design feature(s) and/ or interlock(s) which provide for the following: Breaker interlocks, permissives, bypasses and cross-ties.

Proposed Question: RO Question #30

Technical Reference(s): TQ-TM-104-740-C001, Rev 7 Electrical Print SS-209-050, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: 642-GLO-11

Question Source: Bank #

Modified Bank # 356975

New

Question History: System Exam 12 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:
KA Match: This question matches the KA because the examinee must know design feature of 1M DC that blocks transfer on an ES.

High Cog: This question is high cog because the examinee must recognize that plant conditions would initiate a 4 psi ES Actuation signal and this signal NOT only blocks auto transfer capabilities but prevents manual transfer as well.

1M DC Panel can be powered from 1A or 1B DC Distribution panel. It has both auto and manual transfer capabilities to one of these power sources.

Choose the correct letter below which makes the following statement TRUE.

On an ____________________________ signal, 1M DC panel ____________________ transfer capabilities are prevented.

A. under voltage, only auto

B. Engineered Safeguards, only auto

C. under voltage, both auto and manual

D. Engineered Safeguards, both auto and manual

Answer: D

# EXAMINATION ANSWER KEY
16-01 SENIOR REACTOR OPERATOR NRC EXAM

**31** **ID: 1717750** **Points: 1.00**

Which ONE of the following describes the electrical power supplies to the following pumps associate with Emergency Diesel Generator, EG-Y-1B:
(1) the Fuel Oil Transfer Pumps, DF-P-1C/D, AND
(2) the Auxiliary Fuel Oil Pump, EG-P-10B

A. (1) The two fuel oil transfer pumps are both AC powered.
(2) The Auxiliary Fuel Oil Pump is AC powered.

B. (1) The two fuel oil transfer pumps are both AC powered.
(2) The Auxiliary Fuel Oil Pump is DC powered.

C. (1) One fuel oil transfer pumps is AC powered, One fuel oil transfer pumps is DC powered.
(2) The Auxiliary Fuel Oil Pump is AC powered.

D. (1) One fuel oil transfer pumps is AC powered, One fuel oil transfer pumps is DC powered.
(2) The Auxiliary Fuel Oil Pump is DC powered.

Answer: D

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) Each emergency diesel has two fuel transfer pumps, one is AC and one is DC powered; (2) In addition,each diesel has two fuel oil pumps: One is engine driven and the aux fuel oil pump is DC powered.

| | | |
|---|---|---|
| A. | (1) The two fuel oil transfer pumps are both AC powered. (2) The Auxiliary Fuel Oil Pump is AC powered. | INCORRECT: Plausible since the examinee may believe that AC power is used for both fuel oil transfer pumps in addition to the Auxiliary Fuel Oil Pump. |
| B. | (1) The two fuel oil transfer pumps are both AC powered. (2) The Auxiliary Fuel Oil Pump is DC powered. | INCORRECT: Plausible since the examinee may be believe that both fuel oil transfer pumps are both AC powered and the Auxiliary Fuel Oil Pump is DC powered. |
| C. | (1) One fuel oil transfer pumps is AC powered, One fuel oil transfer pumps is DC powered. (2) The Auxiliary Fuel Oil Pump is AC powered. | INCORRECT: Plausible because the examinee may believe that Auxiliary Fuel Oil Pump is AC powered. |
| D. | (1) One fuel oil transfer pumps is AC powered, One fuel oil transfer pumps is DC powered. (2) The Auxiliary Fuel Oil Pump is DC powered. | CORRECT: According to TQ-TM-104-861, 1A(1B) ES MCC provides power to Diesel Generator A(B) AC Fuel Pump, DF-P-1A (DF-P-1C); while 125/250 VDC Distribution Panel 1P (1Q) provide power to both the DC Fuel Transfer Pump, DF-P-1B (DF-P-1D), and the DC Auxiliary Fuel Pump (EG-P-10A/B). |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 064 | K2.02 |
| | Importance Rating | 2.8 | 3.1 |

K/A: Emergency Diesel Generators: Knowledge of bus power supplies to the following: Fuel Oil Pumps

Proposed Question: RO Question # 31

Technical Reference(s): 1107-5, Rev 151

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-GLO-4

Question Source: Bank # 860277

Modified Bank #

New

Question History: System Exam 9 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments: (KA Match, why high cog, why SRO only)
The KA is matched because the examinee must demonstrate knowledge of bus power supplies to the Fuel oil transfer pumps and the Auxiliary Fuel Oil Pump for a given EDG.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**32** **ID: 1718191** **Points: 1.00**

Plant conditions:

- Loss of Station Power
- EG-Y-1A, Emergency Diesel Generator, had to be started from the Control Room using the Manual Start Pushbutton.
- EG-Y-1A 'Ready to Load' light is DE-ENERGIZED.
- EG-Y-1A voltage is 4050 V.
- EG-Y-1A frequency is 60.5 Hz.

All control room controls associated with EG-Y-1A were in the ES STANDBY line-up when the diesel was started.

From the list below, identify the ONE (1) action that would ENERGIZE the Ready to Load light under these conditions.

A. Adjust the governor to lower frequency to 60.0 Hz.

B. Energize the synchroscope for EG-Y-1A generator output breaker, G1-02.

C. Adjust the local unit voltage rheostat to obtain an output voltage of 4100 V.

D. Adjust the manual voltage controller on Console Right to obtain an output voltage of 4100 V.

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) The 'Ready to Load' light energizes when the diesel is up to the correct operating bands: Voltage is 4100- 4300 volts, and frequency 60.24-61.0 Hz (OP-TM-642-231); (2) The examinee must realize that the voltage is low, but the frequency is in band; (3) The examinee must also recognize that there is only one method to return the voltage to specification, which is the local unit voltage rheostat.

| | | |
|---|---|---|
| A. | Adjust the governor to lower frequency to 60.0 Hz. | INCORRECT: Plausible if the examinee believes that a frequency of 60.5 Hz is out of tolerance for the 'Ready to Load' light. Incorrect because the 'Ready to Load' light will light at 60.24 Hz to 61.0 Hz. |
| B. | Energize the synchroscope for EG-Y-1A generator output breaker, G1-02. | INCORRECT: Plausible if examinee relates a manual start in the control room to a manual start when paralleling an emergency diesel with offsite power which requires the use of the synchroscope. When manually starting a diesel generator to parallel with offsite power, the synchroscope must be energized. For this situation, since there is no offsite power, the synchroscope does not need to be energized. |
| C. | Adjust the local unit voltage rheostat to obtain an output voltage of 4.1KV | CORRECT: See above |
| D. | Adjust the manual voltage controller on Console Right to obtain an output voltage of 4.1 KV. | INCORRECT: Plausible because there is a rheostat for voltage control on console right. Incorrect because this would adjust voltage if the voltage regulator was in manual, which it is not. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 2 | |
| Group # | 1 | |
| K/A # | 064 | A3.03 |
| Importance Rating | 3.4 | 3.3 |

K/A: Emergency Diesel Generators: Ability to monitor automatic operation of the ED/G system, including: Indicating Lights, meters, and recorders.

Proposed Question: RO Question # 32

Technical Reference(s): TQ-TM-104-861-C001, Rev 11

OP-TM-861-901, Rev 18

Proposed References to be provided to applicants during examination: None

Learning Objective: 861-GLO-4

Question Source: Bank # 657377

Modified Bank #

New

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question History: | Sim Exam 2 | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments: (KA Match, why high cog, why SRO only)
KA Match: This question matches the KA because the examinee must know which emergency diesel generator parameters are monitored by the 'Ready-to-Load' light.

**33** **ID: 1718267** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- OP-TM-AOP-005, RIVER WATER SYSTEM FAILURES, is entered due to reduced river water level and rising river water temperature.
- The crew commenced a plant shutdown.

EVENT:

- Auxiliary Operator report:
    - ISPH Pump Bay Water Level is at 270 foot elevation.
    - River Water temperature is 92 degrees F.

Based on these conditions identify the ONE selection that describes required action(s).

A. Continue the shutdown to HOT SHUTDOWN.

B. Continue the shutdown to COLD SHUTDOWN.

C. Trip the reactor and initiate OP-TM-EOP-001, REACTOR TRIP, only.

D. Trip the reactor, initiate OP-TM-EOP-001, REACTOR TRIP, then trip all four Reactor Coolant Pumps.

Answer: D

**Answer Explanation**

<<<<Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-AOP-005, for drought conditions there are separate actions taken for low river water level, high river water temperature, and no nuclear river water and secondary river water pumps operable; (2) Of the two parameters in the stem, only river water elevation is low out of specification; (3) When level was less than 274' a plant shutdown would have been performed to be in HSD (IAW TS 3.0.1); (4) At less than 271', the plant is tripped and all reactor coolant pumps are tripped.

| | | |
|---|---|---|
| A. | Continue the shutdown to HOT SHUTDOWN. | INCORRECT: Plausible since this action is currently being taken due to river water leve < 274 feet. Incorrect because reactor trip criteria has been met. |
| B. | Continue the shutdown to COLD SHUTDOWN. | INCORRECT: Plausible since this is a requirement if river water temperature is > 95F. Incorrect because the stem says river was temperature is 92 F. |
| C. | Trip the reactor and initiate OP-TM-EOP-001, REACTOR TRIP, only. | INCORRECT: Plausible since OP-TM-AOP-005 directs this step when river water level is less than 271'. Incorrect because this answer is incomplete. The procedure directs tripping of all Reactor Coolant pumps as well. |
| D. | Trip the reactor, initiate EOP-001, then trip all four Reactor Coolant Pumps. | CORRECT: IAAT <u>all</u> NR and SR pumps are inoperable or ISPH pump bay water level < 271', then trip the reactor and all 4 reactor coolant pumps. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 076 | A2.01 |
| | Importance Rating | 3.5 | 3.7 |

K/A: Service Water System: 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

Proposed Question: RO Question # 33

Technical Reference(s): OP-TM-AOP-005, Rev 12

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO-7

Question Source: Bank # 355015

Modified Bank #

New

Question History: System Exam 7 Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | b.5 | |
| | 55.43 | | |

Comments: (KA Match, why high cog, why SRO only)
KA Match: This question matches the KA because the examinee must know the procedure actions associated with a low river water level.

**34** **ID: 1718269** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- Main Instrument Air Compressor, IA-P-4, has been secured for repairs.
- Instrument Air Compressors, IA-P-1A and IA-P-1B, are running as required.

EVENTS:

- LOCA
- Loss of Off-Site Power
- 1600 PSI ES Actuation

10 minutes later:

- Due to low instrument air pressure, the CRO starts IA-P-1A.

Cooling for IA-P-1A/B will be aligned to __(1)__ due to the __(2)__.

A. (1) Fire Service Water, (2) ES Actuation Signal.

B. (1) Fire Service Water, (2) Loss of Offsite Power.

C. (1) Secondary Closed Cooling, (2) ES Actuation Signal.

D. (1) Secondary Closed Cooling, (2) Loss of Offsite Power.

Answer: B

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) The normal cooling water supply to the instrument air compressors is Secondary Services Closed Cooling Water system; (2) If none of the closed cooling water pumps are operating (as sensed by the SC pumps breaker position) the three way valves SC-V-57 (IA-P-1A) and SC-V-58 (IA-P-1B) will automatically position to allow Fire Service Water to cool the respective air compressor; (3) Loss of Offsite Power trips the secondary closed cooling pumps, which aligns fire service water to IA-P-1A/B (4) The ES actuation signal effects the auto start of IA-P-1A/B but not the cooling water.

| | | |
|---|---|---|
| A. | (1) Fire Service Water, (2) ES Actuation Signal | INCORRECT: Plausible since Fire Service is the alternate cooling water. If the examinee believes an ES could affect the SCCW Pumps (Load Shed) then the backup cooling would be initiated. |
| B. | (1) Fire Service Water, (2) Loss of Offsite Power | CORRECT: See above |
| C. | (1) Secondary Closed Cooling, (2) ES Actuation Signal | INCORRECT: Plausible if the examinee thinks the backup SCCW pump would start once the emergency diesels started and loaded on their bus. Incorrect because the power supplies to the SCCW pumps are 1C, 1J, and 1N 480V busses. |
| D. | (1) Secondary Closed Cooling, (2) Loss of Offsite Power. | INCORRECT: Plausible if the examinee thinks the backup SCCW Pump would start once the emergency diesel generators started and loaded on their bus. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 2 | |
| Group # | 1 | |
| K/A # | 078 | K1.04 |
| Importance Rating | 2.6 | 2.9 |

K/A: Instrument Air System: Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Cooling water to compressor

Proposed Question: RO Question # 34

Technical Reference(s): 1104-12, Rev 63 TQ-TM-104-850-C001, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-GLO-10

Question Source: Bank #

Modified Bank # 978699

New

Question History: N/A Last NRC Exam: Unmodified on ILT 12-01

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.2 | |
| | 55.43 | | |

Comments: (KA Match, why high cog, why SRO only)
KA Match: This question matches the KA because the examinee must know the cooling water associated with each IA-P-1, and when it each cooling water is utilized.

High Cog: This question is High Cog because the examinee must analyze the events in the stem to determine the cooling water that IA-P-1A/B are utilizing.

Plant Conditions:

The plant is operating at 100% power.
Main Instrument Air Compressor, IA-P-4, has been secured for repairs.
"A" and "B" Instrument Air Compressors, IA-P-1A and IA-P-1B, are running as required.

Event:

A loss of all Secondary Closed Cooling Water Pumps has occurred.

Given the above information and two minutes after the event, the Instrument Air Compressors will align to ___(1)___ as the source of cooling due to ___(2)___.

A. (1) Fire Service water
(2) high Instrument Air compressor temperature

B. (1) Fire Service water
(2) the tripping of all 3 Secondary Closed Cooling Water Pump breakers

C. (1) Nuclear Service Closed Cooling Water
(2) high Instrument Air compressor temperature

D. (1) Nuclear Service Closed Cooling Water
(2) the tripping of all 3 Secondary Closed Cooling Water Pump breakers

Answer: B

**35** **ID: 1718430** **Points: 1.00**

Given the following plant conditions:

- Reactor is at 52% power.
- ICS SG/RX Hand/Auto Station is in HAND.
- All other ICS stations are in AUTOMATIC.

EVENT:
- Group 7 Rod 4 in Quadrant YZ drops fully into the core.

Which of the following indicates the effect on (1) Quadrant Power Tilt in Quadrant YZ and (2) $T_{ave}$ parameter response?

**ASSUME NO OPERATOR ACTIONS.**

A. (1) Negative Quadrant Power Tilt.
(2) $T_{ave}$ lowers and remains low.

B. (1) Negative Quadrant Power Tilt.
(2) $T_{ave}$ lowers and returns to setpoint.

C. (1) Positive Quadrant Power Tilt.
(2) $T_{ave}$ lowers and remains low.

D. (1) Positive Quadrant Power Tilt.
(2) $T_{ave}$ lowers and returns to setpoint.

Answer: B

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) A dropped rod in any quadrant suppress power in that quadrant, thus creating a negative quadrant power tilt; (2) The Dropped Rod Runback is to 55% Reactor Power, so NO runback would be in effect for the given plant conditions; (3) On a Dropped Rod the CRD Out Inhibit clears when Reactor Power is <60% to allow for Automatic Tavg control; (4) In this case, the controlling ICS signal in which it bases the dropped rod runback for is Generated Megawatts, so when rods pull to maintain Tave, power will rise, but the runback will not come back in because Generated Megawatts remain constant; (4) The SG/RX Master in Hand would not effect Tavg Control.

| | | |
|---|---|---|
| A. | (1) Negative Quadrant Power Tilt.<br>(2) Tave lowers and remains low. | INCORRECT: Plausible since part 1 is correct and Tave is low. The CRD Control System would allow Rods to Withdraw to correct for a low Tave, even with a Dropped Rod, as long as the reactor power remains <60%. The examinee may think 55% Reactor Power is the max power since this is the setpoint for a dropped rod runback. (i.e., that control rods will not withdraw); (3) The examinee may assume with the SG/RX Master in Hand it would require manual operation to raise Tave. |
| B. | (1) Negative Quadrant Power Tilt.<br>(2) Tave lowers and returns to setpoint. | CORRECT: The dropped rod would suppress flux in this quadrant and this quadrant would be lower than the other quadrants. On a dropped rod the CRD Out Inhibit clears when Reactor Power is <60% to allow for Automatic Tave control. |
| C. | (1) Positive Quadrant Power Tilt.<br>(2) Tave lowers and remains low. | INCORRECT: Plausible if the examinee is believes that a dropped control rod will give a positive power tilt, based on the calculations. Reversing the operation in the calculation gives the same numerical value but will reverse the sign. The examinee may think 55% Reactor Power is the max power since this is the setpoint for a dropped rod runback so control rods could not withdraw. The examinee may assume with the SG/RX Master in Hand it would require manual operation to raise Tave. |
| D. | (1) Positive Quadrant Power Tilt.<br>(2) Tave lowers and returns to setpoint. | INCORRECT: Plausible if the examinee is believes that a dropped control rod will give a positive power tilt, based on the calculations. Reversing the operation in the calculation gives the same numerical value but will reverse the sign. On a Dropped Rod the CRD Out Inhibit clears when Reactor Power is <60% to allow for Automatic Tavg control |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 2 | | |
| | K/A # | 001 | K3.02 | |
| | Importance Rating | 3.4 | | 3.5 |

K/A: Control Rod Drive System: Knowledge of the effect that a loss or malfunction of the CRDS will have on the following: RCS

Proposed Question: RO Question # 35

Technical Reference(s): OP-TM-AOP-062, Rev 7 TQ-TM-104-621-C001, Rev 10

TQ-TM-104-622-C001, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-062-PCO-5

Question Source: Bank #

Modified Bank # 887495

New

Question History: N/A Last NRC Exam: 2003 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments: (KA Match, why high cog, why SRO only)
KA Match: This question matches the KA because the examinee will have to know which area of the core has a negative quadrant power tilt, in addition to what Tave will do.

High Cog: This question is high cog because the examinee will have to analyze he dropped ro to determine that quadrant power tilt in that region is negative. In addition, the examinee must know the ICS control on an dropped rod, and that there is no CRD out inhibit preventing ICS from maintaining Tave in the correct band.

Sequence of events:

- Reactor power is 100%, with ICS in full automatic.
- Group 7 Rod #4 drops into the core, Quadrant YZ.

Based on these conditions, identify the ONE statement below that describes initial plant response when the rod reaches full insertion.

A. Pressurizer level rises.

B. RCS pressure and temperature lower.

C. Diamond rod control transfers to manual.

D. Quadrant power tilt becomes more positive in Quadrant YZ.

Answer: B

**36** **ID: 1737551** **Points: 1.00**

Plant Conditions:

- 60% power with ICS in auto.

EVENT:

- Reactor Building pressure starts to rise.
- RCS pressure starts to lower.
- The reactor is tripped and the IMA's of OP-TM-EOP-001, REACTOR TRIP are performed.

POST TRIP CONDITIONS:

- Reactor Building pressure rises to 4.2 psig.
- RCS pressure lowers to 1675 psig.

Post Trip, RM-A-2, Reactor Building Atmosphere Monitor, __(1)__ be used to determine whether the Reactor Building pressure increase is from the reactor coolant leak because __(2)__.

A. (1) can
(2) there is NOT a 1600 psig ES Actuation Signal

B. (1) can
(2) RM-A-2 is environmentally qualified for accident conditions

C. (1) can NOT
(2) it is isolated from Containment

D. (1) can NOT
(2) RM-A-2 is NOT environmentally qualified for accident conditions

Answer: C

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) ES Actuation signals, 1600 psig, 500 psig, and 4 psig RB pressure isolate RM-A-2 from the reactor building; (2) Since the stem never indicates that the 1600 psig was bypassed, when RCS pressure lowered below 1600 psig, RM-A-2 is isolated and cannot be used to differentiate between an RCS leak or steam leak; (3) In addition, RM-A-2 pump will de-energize when a combination of its' isolation valves close (CM-V-1, 2, 3, 4).

| | | |
|---|---|---|
| A. | (1) can<br>(2) there is NOT a 1600 psig ES Actuation Signal | INCORRECT: Plausible if the examinee believes that only a 1600 psig ES signal isolates RM-A-2. The 1600 psig signal is indicative of a LOCA, the examinee may believe that in the case of a LOCA that RM-A-2 gets isolated. |
| B. | (1) can<br>(2) Reactor Building Atmospheric conditions will NOT block flow | INCORRECT: Plausible if the examinee does not realize that a 1600 psig signal is present and/or believes that a 4 psig ES actuation is the only ES signal that isolates RM-A-2. |
| C. | (1) can NOT<br>(2) it is isolated from Containment | CORRECT: A 1600 psig ES isolates RM-A-2 via CM-V-1,2,3,4 closing. Also, there is an interlock which will automatically de-energize RM-A-2's pump when either CM-V-1 or CM-V-2 close. |
| D. | (1) can NOT<br>(2) Reactor Building Atmospheric conditions may prevent proper air flow | INCORRECT: Part 1 is correct but part 2 is incorrect. RM-A-2 does have charcoal and paper filters. Plausible if the candidate recognizes that they could saturate if the detector was not isolated but this would not happen since it is Isolated from RB Atmospheric conditions. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 002 | A3.01 |
| | Importance Rating | 3.7 | 3.9 |

K/A: Reactor Coolant System: Ability to monitor automatic operation of the RCS, including: Reactor coolant leak detection system

Proposed Question: RO Question # 36

Technical Reference(s): 1105-3, Rev 053 TQ-TM-104-661-C001, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-2

Question Source: Bank #

Modified Bank #

New X

| | | | |
|---|---|---|---|
| Question History: N/A | | Last NRC Exam: N/A | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments: (KA Match, why high cog, why SRO only)
KA Match: This question matches the KA because the examinee must know whether RM-A-2 is capable of being used for RCS leak detection.

High Cog: This question is high cog because the examinee must determine that a 1600 psig ES signal is present. Then the examinee must determine that on a 1600 psig ES, the containment isolation valves for RM-A-2 go closed.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**37** **ID: 1718596** **Points: 1.00**

Sequence of Events:

- Emergency Feedwater Pump, EF-P-2A, is OOS.
- Loss of Offsite Power (LOOP) occurred 10 minutes ago.
- A small break LOCA occurred following the loss of offsite power.
- Emergency Feedwater Pump, EF-P-2B, tripped when it started automatically.
- OP-TM-EOP-001, Reactor Trip, IMAs were performed.
- OP-TM-EOP-006, LOCA Cooldown and OP-TM-AOP-020, Loss of Station Power, actions are in progress.
- 4 psig ESAS was manually actuated.
- RCS temperature is 518°F and SCM has been maintained >25°F throughout the transient.
- Incore Thermocouples are slowly rising.
- Natural Circulation cannot be verified.

Which ONE of the following is the **NEXT** required action?
Reduce OTSG pressure __________________________.

A. while maintaining >750 psig to avoid feedwater isolation.

B. in both OTSGs as low as possible (Atmospheric pressure or vacuum).

C. in both OTSGs to approximately 150 psig at the maximum cooldown rate allowed.

D. so that secondary Tsat is 40 to 60°F lower than incore thermocouple temperature, while maintaining >150 psig.

Answer: D

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-EOP-006, Section 4.0, INADEQUATE RCS COOLDOWN is entered under the following conditions: Primary to secondary heat transfer cannot be established, RCS temperature is > 300F, and core cooldown rate is less than 40F/hr; (2) Due to the loss of offsite power, there are no reactor coolant pumps. In addition the stem states that natural circulation is not verified. In order to have primary to secondary heat transfer, a reactor coolant pump must be running or natural circulation must be verified; (3) In section 4, the crew will lower OTSG pressures so Tsat is 40 to 60F lower than incore thermocouple temperature in order to maintain the OTSGs as a heat sink; (3) With only the steam driven Emergency Feedwater Pump available (EF-P-1), OTSG pressure must be maintained >150 psig to ensure feed flow to the OTSGs.

| | | |
|---|---|---|
| A. | while maintaining >750 psig to avoid feedwater isolation. | INCORRECT: Plausible since lowering OTSG is required and maintain OTSG Pressure > 750 psig would prevent main Feedwater Isolation, allowing it be available if power is restored. However, the procedure directs bypassing the MFW Isolation signal prior to OTSG pressure < 750 psig. |
| B. | in both OTSGs as low as possible (Atmospheric pressure or vacuum). | INCORRECT: Plausible because this is how the operating crew would perform a rapid RCS cooldown if SCM were lost and an RCP were still running. Incorrect because those symptoms to not exist. |
| C. | in both OTSGs to approximately 150 psig at the maximum cooldown rate allowed. | INCORRECT: Plausible since this would strengthen the heat sink. Incorrect because lowering OTSG pressure rapidly to 150 psig could cause the OTSGs to become uncoupled from the primary and further disrupt the attempts as establishing primary to secondary heat transfer. In addition, the cooldown rate is controlled to lower cold leg temperatures in a slower and more controlled manner. |
| D. | so that secondary Tsat is 40 to 60°F lower than incore thermocouple temperature, while maintaining >150 psig. | CORRECT: See above |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 2 | | |
| | K/A # | 035 | A2.06 | |
| | Importance Rating | 4.5 | | 4.6 |

K/A: Steam Generator System: Ability to (a) predict the impacts of the following malfunctions or operations on the SG; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Small break LOCA

Proposed Question: RO Question # 37

Technical Reference(s): OP-TM-EOP-006, Rev 12

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP004-PCO-3

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question Source: | Bank # | | |
| | Modified Bank # | 572835 | |
| | New | | |
| Question History: | Simulator Exam 9 | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.5 | |
| | 55.43 | | |

Comments: (KA Match, why high cog, why SRO only)
KA Match: This question matches the KA because the examinee must predict the impact that having only the steam drive emergency feedpump has on a LOCA Cooldown. In addition, the examinees must have knowledge of the LOCA cooldown procedure to understand the bands that the operating crew would set to establish the OTSG has a heat sink.

High Cog: This question is high cog because the examinee must ascertain all the the requirements to enter section 4 of OP-TM-EOP-006 exist, and then understand steps within that section to strengthen the OTSG as a heat sink.

Sequence of Events:

- Emergency Feedwater Pump, EF-P-2A, is OOS.
- Loss of Offsite Power (LOOP) occurred 10 minutes ago.
- A small break LOCA occurred following the loss of offsite power.
- Emergency Feedwater Pump, EF-P-2B, tripped when it started automatically.
- OP-TM-EOP-001, Reactor Trip, IMAs were performed.
- OP-TM-EOP-006, LOCA Cooldown and OP-TM-AOP-020, Loss of Station Power, actions are in progress.
- RCS temperature is 518°F and SCM has been maintained >25°F throughout the transient.
- Cooldown rate is 5°F/hr.
- Conditions to verify Natural Circulation can not be met.

Which ONE of the following is the **NEXT** required action?

A. Lower OTSG Pressure while maintaining >750 psig to avoid feedwater isolation in accordance with EOP-006, LOCA Cooldown.

B. Lower OTSG pressure to establish >40°F/hr cooldown rate while maintaining >150 psig for Emergency Feedwater Pump, EF-P-1, operation in accordance with EOP-006, LOCA Cooldown.

C. Lower OTSG Pressure in both OTSG's as low as possible (Atmospheric pressure or vacuum) to promote primary to secondary heat transfer in accordance with EOP-006, LOCA Cooldown.

| | |
|---|---|
| D. | Lower OTSG Pressure so secondary Tsat is 40-60°F higher than incore thermocouple temperature to promote primary to secondary heat transfer in accordance with EOP-004, Lack of Primary to Secondary Heat Transfer. |
| | Answer: B |

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**38** **ID: 1718720** **Points: 1.00**

Plant conditions:

- 100% power with ICS in full auto.

EVENT:

- Overhead alarm MAP C-1-1, "RADIATION LEVEL HI", just actuated.
- The source of the alarm is RM-A-5, CONDENSER VACUUM PUMP EXHAUST.

The MAP-5 Sampler starts to sample the:

A. steam lines for only iodine when RM-A-5 reaches the ALERT setpoint.

B. steam lines for iodine and tritium when RM-A-5 reaches the ALERT setpoint.

C. condenser offgas for only iodine when RM-A-5 reaches the HI-ALARM setpoint.

D. condenser offgas for iodine and tritium when RM-A-5 reaches the HI-ALARM setpoint.

Answer: C

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) The initiating setpoint for starting the MAP-5 is from the HI Alarm NOT the Alert Setting; (2) Physical location of the sample line is from Condenser Offgas, NOT from the Steam Lines; (3) The MAP-5 detects Iodine NOT tritium.

| | | |
|---|---|---|
| A. | steam lines for only iodine when RM-A-5 reaches the ALERT setpoint. | INCORRECT: Plausible since both OTSGs have radiation monitors on the steam lines to detect for an OTSG Tube Leak. Incorrect because this is not the location of the interface for RM-A-5. In addition, the alert alarm does NOT start the MAP-5. |
| B. | steam lines for iodine and tritium when RM-A-5 reaches the ALERT setpoint. | INCORRECT: Plausible since both OTSGs have radiation monitors on the steam lines and there is a parallel line for a tritium sample for the MAP-5. In addition, the Alert Alarm does NOT start the MAP-5. |
| C. | condenser offgas for only iodine when RM-A-5 reaches the HI-ALARM setpoint | CORRECT: See above. |
| D. | condenser offgas for iodine and tritium when RM-A-5 reaches the HI-ALARM setpoint. | INCORRECT: Plausible since it does sample for Iodine, however the sample line for Tritium is in parallel and a manual operation not associated with the MAP-5. |

Examination Outline Cross-reference:

| | RO | SRO |
|---|---|---|
| Level | RO | SRO |
| Tier # | 2 | |
| Group # | 2 | |
| K/A # | 055 | K1.06 |
| Importance Rating | 2.6 | 2.6 |

K/A: Condenser Air Removal System: Knowledge of the physical connections and/or cause/effect relationships between the CARS and the following systems: PRM System

Proposed Question: RO Question # 38

Technical Reference(s): TQ-TM-104-661-C001 OP-TM-MAP-C0101, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-9

Question Source: Bank # 575075

Modified Bank #

New

Question History: System Exam 11 Last NRC Exam: ILT 05-1

Question Cognitive Level: Memory or Fundamental Knowledge X

| | | |
|---|---|---|
| | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.41 | b.7 |
| | 55.43 | |

Comments: (KA Match, why high cog, why SRO only)
KA Match: This question matches the KA because the examinee must have knowledge of where the radiation monitor connects to the condenser air removal system.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**39** **ID: 1740782** **Points: 1.00**

Given the following valves:

- IA-P-1A/B: INSTRUMENT AIR COMPRESSOR "A"/"B".
- IA-V-1: SERVICE AIR TO IA X-CONNECT.
- IA-V-26: SECONDARY PLANT IA SUPPLY VALVE.
- IA-V-2104A/B: IA-P-1A/B SUPPLY TO IA SYS BLOCK VALVE.
- IA-V-2133: IA-Q-2 (DRYER) & IA-F-10A/B (POST FILTER) BYPASS VALVE.

Which of the following is the correct sequence of events that automatically occur in the Instrument and Service Systems as air pressure lowers?

A. 85 psig IA-V-2104A/B OPEN and IA-P-1A/B START.
80 psig IA-V-1 OPENS.
75 psig IA-V-2133 OPENS.
60 psig IA-V-26 CLOSES.

B. 90 psig IA-V-2133 OPENS.
85 psig IA-V-1 OPENS.
75 psig IA-V-2104A/B OPEN and IA-P-1A/B START.
60 psig IA-V-26 CLOSES.

C. 95 psig IA-V-2133 OPENS.
90 psig IA-V-2104A/B OPEN and IA-P-1A/B START.
85 psig IA-V-26 CLOSES.
70 psig IA-V-1 OPENS.

D. 100 psig IA-V-2104A/B OPEN and IA-P-1A/B START.
90 psig IA-V-1 OPENS.
85 psig IA-V-2133 OPENS.
70 psig IA-V-26 CLOSES.

Answer: A

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) The instrument and station air system setpoints, which are: IA-V-2104A/B open at 85 psig, when this valve opens, IA-P-1A/B will start since pressure is less than 95 psig at PS-465A and B, IA-V-26 Closes at 60 psig, IA-V-2133 will open at 75 psig, IA-V-26 closes at 60 psig.

| | | |
|---|---|---|
| A. | 85 psig IA-V-2104A/B OPEN and IA-P-1A/B START.<br>80 psig IA-V-1 OPENS.<br>75 psig IA-V-2133 OPENS.<br>60 psig IA-V-26 CLOSES. | Correct Answer: See above. |
| B. | 90 psig IA-V-2133 OPENS.<br>85 psig IA-V-1 OPENS.<br>75 psig IA-V-2104A/B OPEN and IA-P-1A/B START.<br>60 psig IA-V-26 CLOSES. | INCORRECT: Plausible if the examinee does not know all of the setpoints associated with Instrument and Service Air. |
| C. | 95 psig IA-V-2133 OPENS.<br>90 psig IA-V-2104A/B OPEN and IA-P-1A/B START.<br>85 psig IA-V-26 CLOSES.<br>70 psig IA-V-1 OPENS. | INCORRECT: Plausible if the examinee does not know all of the setpoints associated with Instrument and Service Air. |
| D. | 100 psig IA-V-2104A/B OPEN and IA-P-1A/B START.<br>90 psig IA-V-1 OPENS.<br>85 psig IA-V-2133 OPENS.<br>70 psig IA-V-26 CLOSES. | INCORRECT: Plausible if the examinee does not know all of the setpoints associated with Instrument and Service Air. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 079 | K4.01 |
| | Importance Rating | 2.9 | 3.2 |

K/A: Station Air System (SAS): Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: Cross-Connect with IAS

Proposed Question: RO Question #39

Technical Reference(s): TQ-TM-104-850-C001, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-GLO-6

Question Source: Bank # 375161

Modified Bank #

New

Question History: Sim Exam 3 Last NRC Exam: 05-01

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments:

KA Match: This question matches the KA because the examinee must know multiple instrument air / station air setpoints, including the setpoint for IA-V-1, Service Air to IA cross-connect.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**40** **ID: 1718755** **Points: 1.00**

REFERENCE PROVIDED

Plant conditions:

- Plant is shutting down to HOT SHUTDOWN.
- Tave is 575 deg-F.
- Pressurizer level is being maintained in accordance with 1102-10, PLANT SHUTDOWN.

Based on the above conditions, which indicated level below would be the **minimum** acceptable pressurizer level?

A. 100".

B. 185".

C. 210".

D. 220".

Answer: B

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) 1102-10 Plant Shutdown pressurizer must be maintained with OP-TM-211-472 Attachment 7.2; (2) Interpret and use attachment 7.2 to determine the minimum level based on the given plant conditions with a Tave of 575 deg-F is 185".

| | | |
|---|---|---|
| A. | 100" | INCORRECT: Plausible if the examinee believes that we should be maintaining Pressurizer level for hot shutdown. |
| B. | 185" | CORRECT: See above. |
| C. | 210" | INCORRECT: Plausible if the examinee uses the recommended vs Tave. |
| D. | 220" | INCORRECT: Plausible if the examinee believes that pressurizer level is to be maintained at 220", the normal operating pressurizer level. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 3 | |
| Group # | 1 | |
| K/A # | 2.1.25 | |
| Importance Rating | 3.9 | 4.2- |

K/A: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: RO Question # 40

Technical Reference(s): 1102-10, Rev 99 OP-TM-211-472, Rev 4

Proposed References to be provided to applicants during examination: OP-TM-211-472, Attachment 7.2 with Blacked out curve names.

Learning Objective: GOP-010-PCO-1

Question Source: Bank # 371611

Modified Bank #

New

Question History: Sim Exam 2 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.10

55.43

Comments: (KA Match, why high cog, why SRO only)
KA Match: This matches the KA because the examinee must be able to use a graph which operators use to cooldown the plant.

High Cog: This question is high cog because the examinee must understand how to read the graph given the plant parameters.

**41** **ID: 1737589** **Points: 1.00**

Plant Conditions:

- The plant was at 100% power when a control rod in Group 7 dropped.
- During the runback two additional rods in Group 7 become stuck at 88% withdrawn.
- The ICS runback was completed in automatic.

With the above plant conditions the operating crew must ____________within one hour.

A. trip the reactor

B. be in HOT SHUTDOWN

C. verify Axial Power Imbalance is within limits

D. verify SDM ≥ 1% delta k/k or initiate boration

Answer: D

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) A dropped rod in group 7 will cause a plant runback to less than 455 MWe (55% reactor power), this will drive in Group 7 control rods to ~ 25% rod index; (2) When the dropped rod runback is complete, the crew must enter OP-TM-AOP-062, INOPERABLE ROD due to the two stuck rods being inoperable (control rods misaligned with the group by more than 9 inches); (3) When two or more rods are inoperable, OP-TM-AOP-062, Step 3.2 requires that a shutdown margin (SDM) calculation be performed to verify SDM ≥ 1% delta k/k or initiate emergency boration.

| | | |
|---|---|---|
| A. | trip the reactor | INCORRECT: Plausible because operation with two or more rods inoperable is not permitted. Incorrect because a reactor trip is not required. |
| B. | be in HOT SHUTDOWN | INCORRECT: Plausible because the operating crew will initiate a plant shutdown to HOT SHUTDOWN. Incorrect because the crew has 6 hours to be in HOT SHUTDOWN. |
| C. | verify Axial Power Imbalance is within limits | INCORRECT: Plausible because there are limits on Axial Power which would require action within an hour, however there is nothing in the stem which would indicate that this limit is being violated. |
| D. | verify SDM ≥ 1% delta k/k or initiate boration | CORRECT: See Above. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 3 | |
| Group # | 2 | |
| K/A # | 2.2.39 | |
| Importance Rating | 3.9 | 4.5 |

K/A: Knowledge of less than or equal to one hour Technical Specification action statements for systems.

Proposed Question: RO Question #41

Technical Reference(s): OP-TM-AOP-062, Rev 7 OP-TM-MAP-H0101, Rev 2
OP-TM-AOP-0621, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-062-PCO-4

Question Source: Bank # 1110401
Modified Bank #
New

Question History: Comp 2 Last NRC Exam: N/A

| | | |
|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | |
| | Comprehension or Analysis | X |
| 10 CFR Part 55 Content: | 55.41 | b.10 |
| | 55.43 | |

Comments:

KA Match: This question matches the KA because the examinee must recognize a condition in which SDM must be verified within an hour. This is an operational 1 hour technical specification.

High Cog: This question is high cog because the examinee must analyze the stem to know that control rods have driven in further than 9 inches more than 88% index. I

**42** **ID: 1718778** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- The Unit has experienced several fuel pin failures.
- A leak must be repaired in the Aux. Bldg.
- The general area dose rate in the location of the repair is 600 mrem/hr.
- In order to reach the location of the repair the worker must transit through a 6 Rem/hr high radiation area for 2 minutes and return via the same path.
- The worker currently has an accumulated annual dose of 400 mrem.

The maximum allowable time that the worker can participate in the repairs and **NOT** exceed the TEDE Administrative Dose Control Limit is _________ minutes (Worker does **NOT** have a High Lifetime Exposure).

A. 100

B. 120

C. 140

D. 160

Answer: B

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

Explanation: To answer this question correctly, the examinee must know: (1) The TEDE administrative dose limit in RP-AA-203 is 2000 mrem routine cumulative TEDE/yr; (2) That the worker in question already has 400 mrem and must make a trip TO and FROM the work area without exceeding the TEDE limit.

| | | |
|---|---|---|
| A. | 100 | INCORRECT: Based on 2000 mrem - 600 mrem (current) - 400 mrem (transient) = 1000 / 600 mrem/hr x 60 min = 100 minutes |
| B. | 120 | B. CORRECT : The candidate should determine that the ADCL is 2000 mrem.<br>Transient exposure is 400 mrem (6000mrem/hr x 4/60hr). (transit to and from the job).<br>(Current) 400 mrem + (transit) 400 mrem = 800 mrem<br>ADCL of 2000 mrem - 800 mrem = 1200 mrem allowable before reaching ADCL.<br>1200 mrem /600 mrem/hr = 2 hours |
| C. | 140 | INCORRECT: Based on calculating using a one-way transit dose |
| D. | 160 | INCORRECT: Based on using ADCL (2000) and NO transit dose. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 3 | |
| Group # | 3 | |
| K/A # | 2.3.4 | |
| Importance Rating | 3.2 | 3.7 |

K/A: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: RO Question # 42

Technical Reference(s): RP-AA-203, Rev 5

Proposed References to be provided to applicants during examination: None

Learning Objective: None

Question Source: Bank # 375097

Modified Bank #

New

Question History: Sim Exam 3 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

| | | |
|---|---|---|
| 10 CFR Part 55 Content: | 55.41 | b.12 |
| | 55.43 | |

Comments:

KA MATCH: This question matches the KA because the examinee must have knowledge of radiation exposure limits.

High Cog: This question is high cog because the examinee must use math to determine the correct allowable time the worker has in the area.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**43** **ID: 1718893** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Sequence of Events:

T = 1100:
A Reactor Building Purge has commenced.

T = 1200:
The Reactor Building purge is secured due to a minor equipment problem.

T = 1730:
The equipment problem has been fixed and the Reactor Building purge is ready to recommence.

Given the above information and IAW CY-TM-170-2012, RELEASING RADIOACTIVE GASEOUS EFFLUENTS:

The RB Purge may recommence________________.

A. with the same release permit using original air sample data.

B. with the same release permit using updated air sample data.

C. when a new release permit is generated using a new release number with updated air sample data.

D. when a new release permit is generated using the same release number with updated air sample data.

Answer: C

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) In accordance with CY-TM-170-2012, step 4.1.3: A Reactor Building Purge may be stopped and restarted within four hours, using the same release permit; (2) If the purge is secured for more than four hours, then INITIATE a new release permit (with **new** air samples).

| | | |
|---|---|---|
| A. | with the same release permit using original air sample data. | INCORRECT: Plausible if the candidate is either not familiar with the time requirement allowed to use the same permit without a new air sample. |
| B. | with the same release permit using updated air sample data. | INCORRECT: Plausible if the candidate is not familiar with the time requirement allowed to use the same permit even with a new air sample. |
| C. | when a new release permit is generated using a new release number with updated air sample data. | CORRECT: See above. |
| D. | when a new release permit is generated using the same release number with updated air sample data. | INCORRECT: Plausible if the candidate thinks the same Permit Number is permitted, since the old permit allowing the purge was not completed. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 3 | |
| | K/A # | 2.3.11 | |
| | Importance Rating | 3.8 | 4.3 |

K/A: Ability to control radiation releases.

Proposed Question: RO Question # 43

Technical Reference(s): CT-TM-170-2012, Rev 1

Proposed References to be provided to applicants during examination: None

Learning Objective: NOP-DBIG-APCO-1

Question Source: Bank #

Modified Bank # 1142262

New

Question History: N/A Last NRC Exam: Unmodified on 12-01 NRC

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.10

55.43

Comments: (KA Match, why high cog, why SRO only)
KA Match: The KA is matched because the question requires the candidates to recognize the requirements prior to reestablishing a Reactor Building Purge.

High Cog: This question is at the Comprehension/Analysis level because the candidate is required to perform a math problem and, based on the results of the math problem, the candidate must recall procedural steps and requirements associated with Reactor Building Purge.

Plant Conditions (T = 1600):

- The plant is operating at 100% power.
- A Reactor Building purge has commenced.

Sequence of Events:

- T = 1700:
  - The Reactor Building purge is secured due to the Chemistry Supervisor questioning the calculations used.
- T = 2030:
  - The Chemistry Supervisor agrees that all calculations are correct and the Reactor Building purge is ready to recommence.

GIven the above information and IAW 6610-ADM-4250.12, Releasing Radioactive Gaseous Effluents, which ONE of the following statements is correct with regards to the Reactor Building purge?

A. The same release permit may be used.

B. No release permit is needed for a Reactor Building purge.

C. A new release permit needs to be generated with a new release number assigned.

D. A new release permit needs to be generated but the same release number can be used.

Answer: A

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**44** **ID: 1719768** **Points: 1.00**

In accordance with OS-24, which of the following is correct regarding the interruption of the Reactor Trip Immediate Manual Actions?

A. If a Loss of Subcooling Margin is identified, Rule 1 must be performed to trip the reactor coolant pumps, initiate HPI and EFW.

B. If an Excessive Heat Transfer is identified, Rule 3 must be performed to isolate the affected OTSG.

C. If a Lack of Heat Transfer is identified, Rule 4 must be performed to establish feedwater control.

D. If multiple dropped rods are identified, Rule 5 must be completed to initiate Emergency Boration.

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, provides direction that if a loss of Subcooling Margin is identified during the performance of the Reactor Trip Immediate Manual Actions, the crew member identifying the symptom announces the symptom to the team, the Rule is pulled, concurrence is obtained from the Control Room Supervisor, and the Rule is immediately performed by the Assistant Reactor Operator; (2) The reason for this is that the Reactor Coolant Pumps must be tripped within one minute from loss of subcooling margin.

| | | |
|---|---|---|
| A. | If a Loss of Subcooling Margin is identified, Rule 1 must be performed to trip the reactor coolant pumps, initiate HPI and EFW. | CORRECT: See above |
| B. | If an Excessive Heat Transfer is identified, Rule 3 must be performed to isolate the affected OTSG. | INCORRECT: Plausible if the examinee believes that Excessive Heat Transfer has a time critical action that is required to be taken immediately. |
| C. | If a Lack of Heat Transfer is identified, Rule 4 must be performed to establish feedwater control. | INCORRECT: Plausible if the examinee believes that Lack of Heat Transfer has a time critical action that is required to be taken immediately |
| D. | If multiple dropped rods are identified, Rule 5 must be completed to initiate Emergency Boration. | INCORRECT: Plausible if the examinee believes that the dropped rods must be addressed immediately. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 4 | |
| | K/A # | 2.4.16 | |
| | Importance Rating | 3.5 | 4.4 |

K/A: Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.

Proposed Question: RO Question # 44

Technical Reference(s): OS-24, Rev 28

Proposed References to be provided to applicants during examination: None

Learning Objective: None

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | |
|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X |
| | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.41 | b.10 |
| | 55.43 | |

Comments:

KA MATCH: This question matches the KA because the examinees must have knowledge of the procedure hierarchy in OS-24.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**45** **ID: 1719786** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- The reactor has tripped.
- Damage has occurred to the Control Tower, preventing plant control from either the Control Room or the Remote Shutdown Panels.
- On-shift personnel are not responding to any communications.

Based on the above conditions, the priority action to be taken by an Auxiliary Operator is to report to the:

A. EFW area and perform OP-TM-AOP-009, Loss of Plant Control Facilities in order to establish EFW flow.

B. Auxiliary Boilers and perform OP-TM-414-401 (402), to start AS-B-1A (1B) in order to establish Auxiliary Steam for necessary loads.

C. Auxiliary Building and perform Attachment 5 of OP-TM-EOP-020, Cooldown From Outside of Control Room to prevent spurious operation of MOV's.

D. Turbine Building and open AS-V-8 IAW Attachment E of OS-24, Conduct of Operations During Abnormal and Emergency Events, to make Auxiliary Steam available to Gland Steam when Auxiliary Steam pressure is available.

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) According to OS-24, attachment E, in the event of damage to the Control Tower preventing plant control from either the Control Room or Remote Shutdown area all available NLO's should report to the EFW area and implement AOP-009; (2) AOP-009 should be performed immediately without CR concurrence; (3) The mitigation strategy for AOP-009 is to establish EFW flow to the OTSGs to remove decay heat.

| | | |
|---|---|---|
| A. | EFW area and perform OP-TM-AOP-009, Loss of Plant Control Facilities in order to establish EFW flow. | CORRECT: See above |
| B. | Auxiliary Boilers and perform OP-TM-414-401 (402), to start AS-B-1A (1B) in order to establish Auxiliary Steam for necessary loads. | INCORRECT: Plausible because this is an assigned post trip action, following a Reactor Trip. Incorrect because OS-24 directs all NLO's to the EFW area to establish feedwater. |
| C. | Auxiliary Building and perform Attachment 5 of OP-TM-EOP-020, Cooldown From Outside of Control Room to prevent spurious operation of MOV's. | INCORRECT: Plausible because the control room is not available. Incorrect because in this case, EOP-020 is not entered because the whole control tower is not available. |
| D. | Turbine Building and open AS-V-8 IAW Attachment E of OS-24, Conduct of Operations During Abnormal and Emergency Events, to make Auxiliary Steam available to Gland Steam when Auxiliary Steam pressure is available. | INCORRECT: Plausible because this is an assigned post trip action, following a Reactor Trip. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 4 | |
| | K/A # | 2.4.35 | |
| | Importance Rating | 3.8 | 4.0 |

K/A: Knowledge of local auxiliary operator tasks during an emergency and the | resultant operational effects.

Proposed Question: RO Question #45

Technical Reference(s): OS-24, Rev 28 OP-TM-AOP-009, Rev 10

Proposed References to be provided to applicants during examination: None

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Learning Objective: | AOP-009-PCO-4 | | |
| Question Source: | Bank # | 862353 | |
| | Modified Bank # | | |
| | New | | |
| Question History: | Sim Exam 9 | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge X | | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must know where the location and actions taken by the AO's on a loss of plant facilities and resultant operational effects.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**46** **ID: 1720267** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- A feedwater transient occurs.
- An automatic reactor trip on HIGH RCS temperature fails to occur.
- Main Feedwater remains available.
- The reactor trip and DSS Pushbuttons fail to trip the reactor.
- A CRO opens both the 1L-02 and 1G-02 breakers.
- Several groups of control rods failed to insert.
- Reactor power is 10% and steady.

Based on the above conditions, the CRO must __(1)__ in order to __(2)__ .

A. (1) trip the Main Turbine,
(2) minimize peak RCS pressure

B. (1) trip the Main Turbine,
(2) prevent an overcooling event

C. (1) wait until RCS Pressure is less than 2500 psig, then initiate Emergency Injection HPI/LPI,
(2) inject borated water to the RCS

D. (1) wait until RCS Pressure is less than 2500 psig, then initiate Emergency Injection HPI/LPI,
(2) prevent RCS pressure from exceeding 3000 psig

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) The reactor did not shutdown with an automatic or manual trip signal, this is an ATWS; (2) When this happens the operator opens 1L-02 and 1G-02, which should remove all power from the CRDMs; (3) After the breakers are open, since multiple groups of rods did not insert, and reactor power is greater than 5%, that indicates that the reactor is still NOT shutdown; (4) Since feedwater is available, the mitigation strategy is to maintain primary to secondary heat transfer and emergency borate with HPI (charging pumps).

| | | |
|---|---|---|
| A. | (1) trip the Main Turbine,<br>(2) minimize peak RCS pressure | INCORRECT: Plausible because if Main Feedwater were not available, then the operator would trip the turbine and initiate emergency feedwater. |
| B. | (1) trip the Main Turbine,<br>(2) prevent an overcooling event | INCORRECT: Plausible because if Main Feedwater were not available, then the operator would trip the turbine and initiate emergency feedwater. |
| C. | (1) wait until RCS Pressure is less than 2500 psig, then initiate Emergency Injection HPI/LPI,<br>(2) inject borated water to the RCS | CORRECT: See above. |
| D. | (1) wait until RCS Pressure is less than 2500 psig, then initiate Emergency Injection HPI/LPI,<br>(2) prevent RCS pressure from exceeding 3000 psig | INCORRECT: Plausible since HPI/LPI is initiated when RCS pressure is < 2500 psig, however preventing the RCS Pressure from exceeding 3000 psig is associated with maintaining the Main Turbine on-line. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 029 | EA1.01 |
| | Importance Rating | 3.4 | 3.1 |

K/A: Anticipated Transient Without Scram (ATWS): Ability to operate and monitor the following as they apply to a ATWS: Charging Pumps

Proposed Question: RO Question # 46

Technical Reference(s): OP-TM-EOP-001, Rev 16 OP-TM-EOP-0011, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-001-PCO-5

Question Source: Bank #

Modified Bank #

| | | | | |
|---|---|---|---|---|
| | New | X | | |
| Question History: | N/A | | Last NRC Exam: | N/A |
| Question Cognitive Level: | | Memory or Fundamental Knowledge | | X |
| | | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | | 55.41 | b.7 | |
| | | 55.43 | | |

Comments:
KA Match: The examinee must know the the mitigation strategy for shutting down the reactor on an ATWS, which involves using the charging pumps to emergency borate.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**47** **ID: 1720289** **Points: 1.00**

A reactor trip from 100% power has occurred.

The following plant conditions exist:

- All four reactor coolant pumps are tripped.
- Emergency Feedwater CANNOT be established to either OTSG.
- Main Feedwater CANNOT be recovered.
- Incore thermocouples are indicating 592°F and rising slowly.
- OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, has been entered.
- SCM is 28°F and lowering.

The CRS must __(1)__ based on __(2)__.

A. (1) go to OP-TM-EOP-009, HPI COOLING, and perform Rule 1 when Subcooling Margin is < 25°F,
(2) SCM approaching 25°F

B. (1) go to OP-TM-EOP-009, HPI COOLING, and perform Rule 1 when Subcooling Margin is < 25°F,
(2) RCS pressure approaching 2450 psig

C. (1) continue with OP-TM-EOP-004 to open the PORV then close it when RCS pressure is 1750 psig,
(2) RCS pressure approaching 2450 psig

D. (1) continue with OP-TM-EOP-004 to open the PORV then close it when SCM approaches 30°F,
(2) SCM approaching 25°F

Answer: A

| **Answer Explanation** |
|---|

<<Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-EOP-004, if SCM is approaching 25°F, the crew will go to OP-TM-EOP-009; (2) Additionally, the examinee must determine this is the correct course of action by using the incore temperature at 592°F and the SCM at 28°F to determine that RCS pressure is approximately 1800 psig (from the steam tables); (3) OP-TM-EOP-009 could be entered in two separate ways from OP-TM-EOP-004: The first was previously discussed when SCM is approaching 25°F, the second is if RCS pressure is approaching 2450 psig with no FEEDWATER available; (4) Due to SCM being 28°F and the examinee determining that pressure is not approaching 2450 psig, entry into OP-TM-EOP-009 is required via the approaching 25°F SCM method.

| | | |
|---|---|---|
| A. | (1) go to OP-TM-EOP-009, HPI COOLING, and perform Rule 1 when Subcooling Margin is < 25°F,<br>(2) SCM approaching 25°F | CORRECT: See above. |
| B | (1) go to OP-TM-EOP-009, HPI COOLING, and perform Rule 1 when Subcooling Margin is < 25°F,<br>(2) RCS pressure approaching 2450 psig | INCORRECT: Plausible because OP-TM-EOP-009 is required, incorrect because it is required because SCM is approaching 25°F, not due to RCS pressure approaching 2450 psig. |
| C. | (1) continue with OP-TM-EOP-004 to open the PORV then close it when RCS pressure is 1750 psig,<br>(2) RCS pressure approaching 2450 psig | INCORRECT: Plausible because OP-TM-EOP-004 provides guidance on RCS pressure approaching 2450 psig, which would be to deep cycle the PORV. Incorrect because this is only done if FEEDWATER is available. |
| D. | (1) continue with OP-TM-EOP-004 to open the PORV then close it when SCM approaches 30°F,<br>(2) SCM approaching 25°F | INCORRECT: Plausible because OP-TM-EOP-004 provides guidance on deep cycling the PORV when RCS pressure is approaching 2450 psig. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 1 | | |
| | Group # | 1 | | |
| | K/A # | E04 | EK1.2 | |
| | Importance Rating | 4.0 | | 4.2 |

K/A: Inadequate Heat Transfer: Knowledge of the operational implications of the following concepts as they apply to the (Inadequate Heat Transfer): Normal, abnormal, and emergency operating procedures associated with (Inadequate Heat Transfer).

Proposed Question: RO Question # 47

Technical Reference(s): OP-TM-EOP-004, Rev 11 OP-TM-EOP-009, Rev 8

Proposed References to be provided to applicants during examination: Steam Table

# EXAMINATION ANSWER KEY
16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Learning Objective: | EOP-004-PCO-5 | | |
| Question Source: | Bank # | 300014 | |
| | Modified Bank # | | |
| | New | | |
| Question History: | N/A | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | b.10 | |

Comments:
KA Match: This question matches the KA because the examinee must to the operational implication of approaching 25°F during a lack of heat transfer. They must have knowledge of the transition to OP-TM-EOP-009.

High Cog: This question is high cog because the examinee must analyze the conditions in the stem or in order to determine the correct step to implement in OP-TM-EOP-004.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**48** **ID: 1720350** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Loss of #8 Bus

Based on the Loss of the #8 Bus, the CRS must enter __(1)__ and the CRO is required to make a plant announcement using the __(2)__.

A. (1) OP-TM-AOP-013, LOSS OF 1D 4160V BUS, (2) "RED" plant page and radio

B. (1) OP-TM-AOP-013, LOSS OF 1D 4160V BUS, (2) "GREY" plant page and radio

C. (1) OP-TM-AOP-014, LOSS OF 1E 4160V BUS, (2) "RED" plant page and radio

D. (1) OP-TM-AOP-014, LOSS OF 1E 4160V BUS, (2) "GREY" plant page and radio

Answer: D

**Answer Explanation**

# EXAMINATION ANSWER KEY



<<Explanation: To answer this question correctly, the examinee must know: (1) When the plant loses the 8 bus, the 1A and 1C 4160V busses fast transfer to the B Auxiliary Transformer. Emergency Diesel Generator, EG-Y-1B starts to power the 1E 4160V bus; (2) This meets the entry criteria for entering OP-TM-AOP-014; (3) The power to the "Grey" plant page comes downstream of the 1D 4160V bus and is not effected, hence the plant page is made over the "Grey" phone; (4) OP-TM-AOP-013 directs the plant page to be made over the "Red" phone if the 1D 4160V bus lost power.

| | | |
|---|---|---|
| A. | (1) OP-TM-AOP-013, LOSS OF 1D 4160V BUS, (2) "RED" plant page and radio | INCORRECT: Plausible if the examinee believes that the 1D 4160V bus was lost. OP-TM-AOP-013 would direct the announcement to be made over the "RED" plant page. |
| B. | (1) OP-TM-AOP-013, LOSS OF 1D 4160V BUS, (2) "GREY" plant page and radio | INCORRECT: Plausible if the examinee believes that the 1D 4160V bus was lost. Although the procedure directs use of the "RED" plant page and radio, when the power was restored to the 1D 4160V bus by EG-Y-1A, the "GREY" plant page would work. |
| C. | (1) OP-TM-AOP-014, LOSS OF 1E 4160V BUS, (2) "RED" plant page and radio | INCORRECT: Plausible if the examinee believes that the "GREY" plant page has lost power. Incorrect because the "GREY" plant page gets power from downstream of the 1D 4160V bus. |
| D. | (1) OP-TM-AOP-014, LOSS OF 1E 4160V BUS, (2) "GREY" plant page and radio | CORRECT: See above |

Examination Outline Cross-reference:

| | | |
|---|---|---|
| Level | RO | SRO |
| Tier # | 1 | |
| Group # | 2 | |
| K/A # | A05 | AA2.1 |
| Importance Rating | 3.5 | 4.2 |

K/A: Emergency Diesel Actuation: Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: RO Question #48

Technical Reference(s): OP-TM-AOP-014, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-014-PCO-6

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.10 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the diesel actuates on undervoltage on a loss of the #8 bus. The examinee must recognize the entry criteria for the abnormal procedure which would govern this actuation.

High Cog: This question is high cog because the examinee must analyze the plant status upon loss of the 8 bus, and understand that the 1E 4160V bus is powered by EG-Y-1B, and this is an entry criteria for OP-TM-AOP-014.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**49** **ID: 1720342** **Points: 1.00**

Plant Conditions:

- 40% power with ICS in auto.

EVENT:

- Reactor power is LOWERING.
- RCS pressure is RISING.
- Main Steam Safety Valves are OPEN.
- Turbine Bypass Valves, MS-V-3A-F, and Atmospheric Dump Valves, MS-V-4A/B, are OPEN.
- Indicating lights on Panel SS-1 are GREEN for the breakers for the Middletown 1092, Jackson 1051 and 500 kV tie lines.
- Indicating lights on Panel SS-1 for the Middletown 1091 breaker switches are GREEN and YELLOW.
- Indicating lights for the Main Generator Breakers are RED.
- Generator load is 49 megawatts.

Which of the following procedures must be entered to mitigate this event?

A. MAP-H0101, ICS RUNBACK

B. 1107-11, TMI GRID OPERATIONS

C. OP-TM-AOP-022, LOAD REJECTION

D. OP-TM-AOP-070, PRIMARY TO SECONDARY HEAT TRANSFER UPSET

Answer: C

**Answer Explanation**

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<<<Explanation: To answer this question correctly, the examinee must know: (1) The stem conditions essentially leave the plant in a conditions where the unit is separated from the grid, where our generator breakers are shut powering our Auxiliary Transformers only; (2) This could be caused by multiple things, but the governing procedure in this case will be OP-TM-AOP-022, LOAD REJECTION.

| | | |
|---|---|---|
| A. | MAP-H0101, ICS Runback | INCORRECT: Plausible because both reactor power and MWe are lowering which is also an indication of a plant runback. Incorrect because no runback signal would exist for this situation. |
| B. | 1107-11, TMI Grid Operations | INCORRECT: Plausible because the purpose of this procedure is to provide guidance on operation of components related to Auxiliary Transformers/Switchyard Equipment and to provide guidance for response to grid related abnormal situations. The given symptoms would indicate a potential problem associated with the grid, but the controlling procedure would be AOP-022. |
| C. | OP-TM-AOP-022, Load Rejection. | CORRECT: See above. |
| D. | OP-TM-AOP-070, PRIMARY TO SECONDARY HEAT TRANSFER UPSET | Incorrect: Plausible because this procedure would give direction to the crew if a non-valid primary to secondary heat transfer upset occured. Incorrect because the plant is running back as expected. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 1 | | |
| | Group # | 1 | | |
| | K/A # | 077 | 2.4.11 | |
| | Importance Rating | 4.0 | | 4.2 |

K/A: Generator Voltage and Electrical Grid Disturbances: Knowledge of abnormal condition procedures

Proposed Question: RO Question # 49

Technical Reference(s): OP-TM-AOP-022, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-022-APCO-2

Question Source: Bank # 462877

Modified Bank #

New

Question History: Sim Exam 6 Last NRC Exam: N/A

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.11 | |
| | 55.43 | | |

Comments:
KA Match: This question matches the KA because the examinee must have knowledge of the abnormal operating procedure entered on a loss of the grid.

High Cog: This question is high cog because the examinee must analyze the stem indications to determine that a load rejection has occurred.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**50** **ID: 1720492** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Event:

- RC3-PR Hot Leg A Narrow Range Channel 1 fails LOW.
- MAP alarm H-3-2, SASS Mismatch is illuminated.
- SASS selector switch for RC3A-PT1 Hot Leg A Narrow Range Channel 1 has the RED and WHITE lights illuminated.
- SASS selector switch for RC3B-PT1 Hot Leg B Narrow Range Channel 1 has only the RED light illuminated.

What is the system response of this failure?

A. All Pressurizer Heater Banks energize.

B. SASS transfers to an alternate channel.

C. One channel of ES actuates on Train 'A' AND Train 'B'.

D. Only SCR controlled banks 1, 2, and 3 Pressurizer Heaters energize.

Answer: A

**Answer Explanation**

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) The range of the Narrow Range pressure instrument is 2500 to 1700 psig, and failing low would be 1700 psig; (2) The Narrow Range pressure instrument, RC3A-PT1, is the normally selected SASS instrument to RC3-PR for RCS pressure control; (3) This output of this SASS channel provides control for, PZR Heater Control (Low Pressure), Spray Valve control, and the PORV high pressure setpoint; (4) The effect of this failing low the Pressurizer Heater Banks energize at 2135 psig (SCR Controlled 1, 2 and 3)) and all heater banks (4 and 5) are energized by 2105.

| | | |
|---|---|---|
| A. | All Pressurizer Heater Banks energize. | CORRECT: See above |
| B. | SASS transfers to an alternate channel. | INCORRECT: Plausible if the examinee does not interpret the RED and WHITE light indications correctly, and determines that SASS has actuated. Incorrect because the WHITE light on RC3B-PT1 would be illuminated. |
| C. | One channel of ES actuates on Train 'A' AND Train 'B'. | INCORRECT: Plausible if the examinee believes this pressure instrument inputs to ESAS in a way which would actuate one channel. Incorrect because these Narrow Range instruments only go as low as 1700 psig, and ESAS actuates on a 1600 psig pressure signal. |
| D. | Only SCR Controlled Banks 1, 2, and 3 Pressurizer Heaters energize. | INCORRECT: Plausible if examinee thinks the instrument input only effects the SCR Controlled Heater Banks 1, 2, and 3. Incorrect because the bistable banks (4 and 5) are will also turn on for this failure. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 010 | K6.01 |
| | Importance Rating | 2.7 | 3.1 |

K/A: Pressurizer Pressure Control System: Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems

Proposed Question: RO Question # 50

Technical Reference(s): TQ-TM-104-624-C001, Rev 4

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-11

Question Source: Bank #

Modified Bank #

New X

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question History: N/A | | Last NRC Exam: N/A | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must know how the controlling Narrow Range Pressure instrument effects plant control.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**51** **ID: 1720498** **Points: 1.00**

Identify the conditions that promote Pool Boiler Condenser Cooling Mode (BCM) of Heat Transfer.

The OTSG thermal center remains __(1)__ the RCP spillover elevation.
RCS liquid level is maintained __(2)__ the secondary level.

A. (1) above
(2) above

B. (1) above
(2) below

C. (1) below
(2) above

D. (1) below
(2) below

Answer: B

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) OP-TM-EOP-002 basis document describes the parameters to promote BCM cooling as: RCS condensation and ECCS injection do not cause the RCS liquid level to increase above the secondary level, and secondary fluid temperature is maintained below the temperature of the steam on the primary side of the OTSG tubes. the secondary liquid level is high enough that the secondary OTSG thermal center remains several feet above the RCP spillover elevation.

| | | |
|---|---|---|
| A. | (1) above<br>(2) above | INCORRECT: First part is correct. Plausible if the examinee believes in order to accomplish part 1 then RCS liquid level must above secondary level. |
| B. | (1) above<br>(2) below | CORRECT: See above |
| C. | (1) below<br>(2) above | INCORRECT: Plausible if the examinee believes the OTSG thermal must be below the heat source to induce BCM. |
| D. | (1) below<br>(2) below | INCORRECT: Plausible if the examinee believes the OTSG thermal must be below the heat source to induce BCM. Second part is correct |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | E03 | EK3.1 |
| | Importance Rating | 3.2 | 3.8 |

K/A: Inadequate Subcooling Margin: Knowledge of the reasons for the following responses as they apply to the (Inadequate Subcooling Margin): 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.

Proposed Question: RO Question #51

Technical Reference(s): OP--TM-EOP-0021, Rev 5

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-002-GLO-9

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | b.5 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must have knowledge for the response of raising OTSG level on a loss of subcooling margin. The operating characteristic in the question is Boiler Condenser Cooling, and the effect in the question is what the temperature and level of OTSG water will do to the RCS parameters.

**52** **ID: 1720537** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- RPS Channel A in MANUAL BYPASS.

EVENT:

- Vital bus "A" (VBA) losses power.

Given the information above, identify the selection below that describes (1) the number of remaining channels required to trip and (2) the degree of redundancy.

A. (1) ONE (2) ONE

B. (1) ONE (2) TWO

C. (1) TWO (2) ONE

D. (1) TWO (2) TWO

Answer: B

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) The MANUAL BYPASS switch on an RPS cabinet is used for testing or if an erroneous trip signal is present; (2) With an RPS channel in MANUAL BYPASS, the number of channels required to trip is TWO with a redundancy of TWO; (3) When VBA is lost, a trip signal is generated, regardless of the position of the MANUAL BYPASS switch; (4) This places the plant in a situation where if ONE additional channel generates a trip signal, the plant would trip; (5) Since RPS channels B,C, and D are still operable, there is a degree of redundancy of TWO additional channels which could provide the trip signal.

| | | |
|---|---|---|
| A. | (1) ONE (2) ONE | INCORRECT: Plausible because this possibility would exist if a power supply other than VBA was lost (i.e, VBB, VBC, or VBD). understood. |
| B. | (1) ONE (2) TWO | CORRECT: See above |
| C. | (1) TWO (2) ONE | INCORRECT: Plausible if the examinee believes that since RPS A is in MANUAL BYPASS, that this prohibits all trip signal to be generated. Incorrect because this trip signal is generated on a loss of power. |
| D. | (1) TWO (2) TWO | INCORRECT: Plausible if the examinee believes that since RPS A is in MANUAL BYPASS, that this prohibits all trip signal to be generated. Incorrect because this trip signal is generated on a loss of power.. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 012 | K6.02 |
| | Importance Rating | 2.9 | 3.1 |

K/A: Reactor Protection System: Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Redundant channels

Proposed Question: RO Question #52

Technical Reference(s): TQ-TM-104-641-C001, Rev 2 T.S Definition 1.4.2, Amd 278

OP-TM-AOP-015, Rev 10

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-6

Question Source: Bank # 568246

Modified Bank #

New

Question History: N/A Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must know the effect that a loss of a vital bus will have on the redundant channels of RPS.

High Cog: This question is high cog because the examinee must analyze the effect of the loss of VBA on the RPS system.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**53** **ID: 1720550** **Points: 1.00**

Initial conditions:

- 100% Reactor power.
- AH-E-9A, PENETRATION COOLING FAN, is in Normal-after-start.
- AH-E-9B, PENETRATION COOLING FAN, is in Normal-after-stop.

Event:

- Reactor Trip occurs.
- 4# ES actuation occurs (both trains).
- HVA-2-1, PENETRATION COOLING AIR TEMP HI, alarms.
- Recorder TR-805, PENETR COOL AIR TEMPERATURE RECORDER, points 1-25, all read >205°F and rising.

What MINIMUM action(s) must be taken to lower penetration temperatures?

A. Start AH-E-9B ONLY.

B. Restart AH-E-9A ONLY.

C. Bypass both 4# ES signals and then start AH-E-9B.

D. Ensure AH-D-89, AH-E-9A/B OUTSIDE AIR INLET, and AH-D-90, AH-E-9A/B TURBINE BLDG INLET, are open.

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) That AH-E-9A/9B trip on the following signals: Fire/Smoke detector alarm (TS 671), High Temperature at 195°F (TS 706A/706B), and AH-E-9A is load shed on any ES actuation; (2) When a component is load shed, it cannot be restarted until the ES Actuation signals are cleared; (3) Since AH-E-9B is not affected by load shed, it can be started immediately.

| | | |
|---|---|---|
| A. | Start AH-E-9B ONLY. | CORRECT: See above. This is the MINIMUM requirement to start a penetration cooling fan. |
| B. | Restart AH-E-9A ONLY. | INCORRECT: Plausible if the examinee believes that AH-E-9B is the load shed fan or does not recall the ES Signal must be cleared to restart fan. |
| C. | Bypass both 4# ES signals and then start AH-E-9B. | INCORRECT: Plausible if the examinee believes that both fans are load shed and the correct course of action was to start the fan that was not running. |
| D. | Ensure AH-D-89, AH-E-9A/B OUTSIDE AIR INLET, and AH-D-90, AH-E-9A/B TURBINE BLDG INLET, are open. | INCORRECT: Plausible if the examinee believes that BOTH the inlet dampers to the AH-E-9's must be open to start an AH-E-9. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 2 | |
| Group # | 1 | |
| K/A # | 022 | K4.01 |
| Importance Rating | 2.5 | 3.0 |

K/A: Containment Cooling System: Knowledge of the CCS design feature(s) and/or interlock(s) which provide for the following: Cooling of containment penetrations.

Proposed Question: Question #53

Technical Reference(s): TQ-TM-104-240-C001, Rev 7 HVA-2-1 (HVA, Rev 58)
1104-16, Rev 24

Proposed References to be provided to applicants during examination: None

Learning Objective: 240-GLO-10

Question Source: Bank # 880836
Modified Bank #
New

Question History: System Exam 9 Last NRC Exam: N/A

| | | |
|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | |
| | Comprehension or Analysis | X |
| 10 CFR Part 55 Content: | 55.41 | b.7 |
| | 55.43 | |

Comments:

KA MATCH: This question matches the KA because the examinee must know the load shed interlock for AH-E-9A to answer the question correctly.

High Cog: This question is high cog because in addition to knowing the interlock with AH-E-9A, the examinee must know the actions to start another penetration cooling fan.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**54** **ID: 1720590** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Due to a fault, the feeder breaker to 1D 4160V bus trips open.
- Emergency Diesel Generator, EG-Y-1A is powering the 1D 4160V bus.

The 1N 480V Bus is __(1)__ because __(2)__.

A. (1) energized
(2) the 1N 480V bus automatically repowers from the 1L 480V bus crosstie

B. (1) energized
(2) the 1N 480V bus is repowers when EG-Y-1A repowers the 1D 4160V bus

C. (1) de-energized
(2) the UV results in a 1N 480V bus trip and lockout

D. (1) de-energized
(2) the UV results in a 1N 480V bus trip with NO lockout

Answer: D

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) One Line Diagram of the BOP & 1E Electrical System Distribution (4160V to 480V busses); (2) Due to the UV on the 1D 4160V bus, the 1N 480V bus trips; (3) When EG-Y-1A is powering the bus, the 1N 480V bus can be repowered.

| | | |
|---|---|---|
| A. | (1) energized<br>(2) the 1N 480V bus automatically repowers from the 1L 480V bus crosstie | INCORRECT: Plausible since the 1N bus can be repowered from the 1L 480V bus crosstie, however this requires operator actions. |
| B. | (1) energized<br>(2) the 1N 480V bus is repowered when EG-Y-1A repowers the 1D 4160V bus | INCORRECT: Plausible because the other 480V feeder breakers stay shut when the 1D 4160V bus loses power. Incorrect because the 1N 480V bus feeder breaker opens on an 1D 4160V bus UV. |
| C. | (1) de-energized<br>(2) the UV results in a 1N 480V bus lockout | INCORRECT: Plausible because the 1N bus is de-energized, but only an UV with an ES actuation would activate the lockout for the 1N 480V bus. |
| D. | (1) de-energized<br>(2) the UV results in a 1N 480V bus trip with NO lockout | CORRECT: See above. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 1 | | |
| | K/A # | 062 | K4.07 | |
| | Importance Rating | 2.7 | | 3.1 |

K/A: AC Electrical Distribution System: Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: One-line diagram of 4kV to 480V distribution, including sources of normal and alternative power.

Proposed Question: RO Question #54

Technical Reference(s): TQ-TM-104-740-C001, Rev 007

Proposed References to be provided to applicants during examination: None

Learning Objective: 740-GLO-10

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

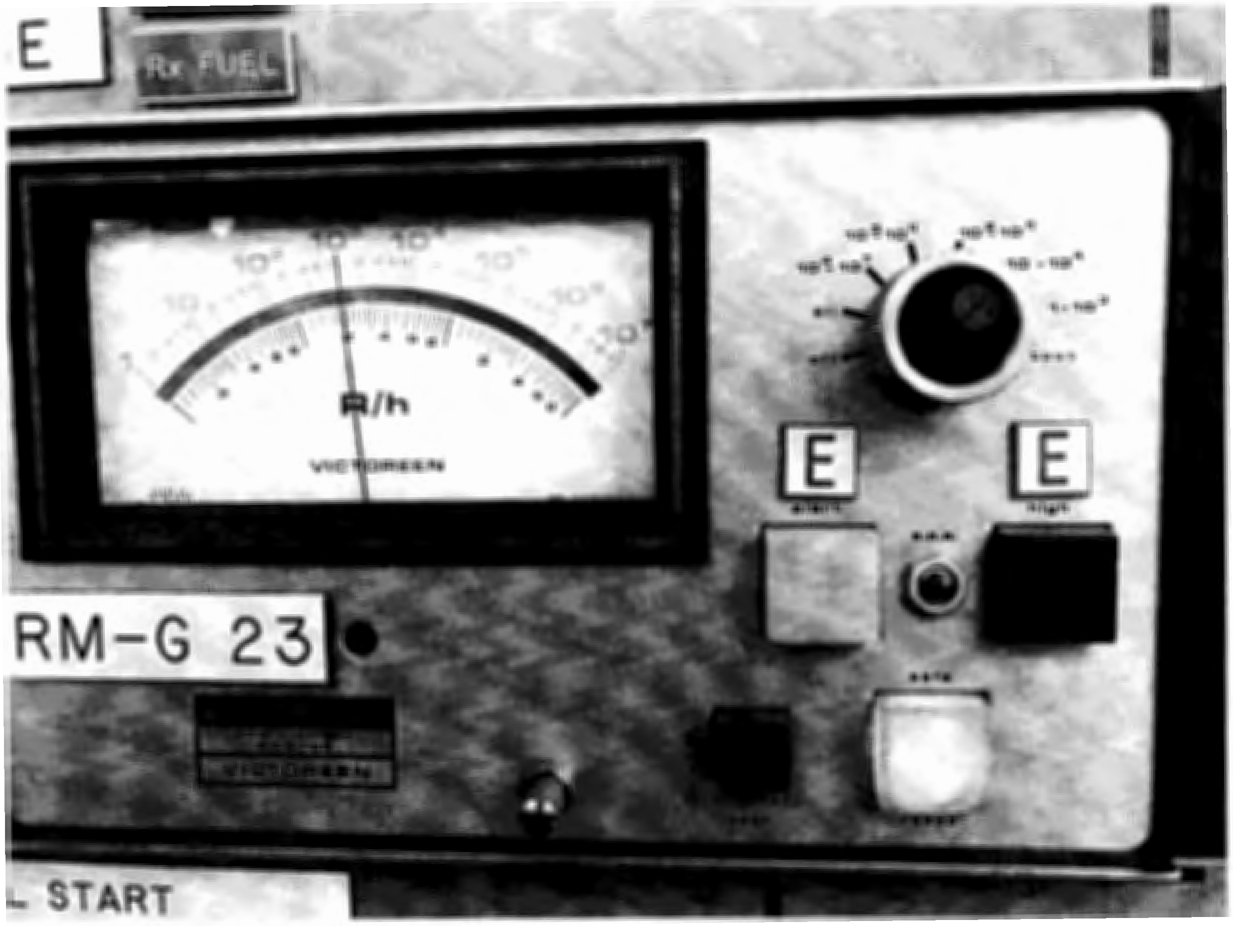
Comments:
KA Match: This question requires knowledge of the one line diagram of the TMI electrical distribution system.

High Cog: This question is high cog because the examinee must know the status of the 1N 480V bus when the associated 4160V bus loses power and is then repowered.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**55** **ID: 1720616** **Points: 1.00**




Identify the correct reading for RM-G-23, RB High Range Radiation Monitor.

A. $1X10^3$ R/h

B. $1X10^3$ mr/h

C. $2X10^2$ R/h

D. $2X10^2$ mr/h

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1)The operation of the indication selector switch for RM-G-23, OFF/ALL/10E7/10E6/10E5/10E4/10E3:
1) Read top scale (RED) in ALL .
2) Read 3 decades in other positions.
Example: Scale Selected 10 - 10E4: Read lower black scale with first mark being 10E1 R/hr and the last mark being 10E4 R/hr
In addition, the black scale does NOT lower the Units to mr/hr.

| | | |
|---|---|---|
| A. | $1X10^3$ R/h | INCORRECT: Plausible if the examinee does correctly interpret the position of the Scale Selector Switch |
| B. | $1X10^3$ mr/h | INCORRECT: Plausible if the examinee does not properly read the scale units. Most Rad Monitors of this type are low range and read out in mr/hr. |
| C. | $2X10^2$ R/h | CORRECT: See above |
| D. | $2X10^2$ mr/h | INCORRECT: Plausible if the examinee misinterprets the black scale as mr/hr. |

Examination Outline Cross-reference:

| | | | |
|---|---|---|---|
| Level | RO | | SRO |
| Tier # | 2 | | |
| Group # | 1 | | |
| K/A # | 073 | 2.1.28 | |
| Importance Rating | 4.1 | | 4.1 |

K/A: Process Radiation Monitoring System: Knowledge of the purpose and function of major system components and controls.

Proposed Question: RO Question #55

Technical Reference(s): TQ-TM-104-661

Proposed References to be provided to applicants during examination: RM-G-23 indication (in color)

Learning Objective: 661-GLO-5

Question Source: Bank #
Modified Bank #
New X

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

| | | | |
|---|---|---|---|
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the function controls for a process radiation monitor.

High Cog: This question is high cog because the examinee must use the picture to determine the correct reading by checking the switch position and the meter reading.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**56** **ID: 1740713** **Points: 1.00**

Plant conditions:

- Reactor at 100% power with ICS in full auto.
- Turbine Bypass Valves for the 'A' OTSG, MS-V-3D, MS-V-3E and MS-V-3F are closed in HAND for ICS module replacement.

EVENT:

- Reactor Trip

The 'A' OTSG will be controlled at __(1)__.

A. 895 psig

B. 965 psig

C. 1010 psig

D. 1026 psig

Answer: D

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) MS-V-3D, E, F are the 'A' OTSG turbine bypass valves; (2) When the ICS module is in hand, they no longer control at the ICS setpoint (which for a reactor trip would be 1010 psig); (3) On a reactor trip, since they will not open at setpoint, the 'A' OTSG will be controlled by the MS-V-4A (Atmospheric Dump Valve) at its setpoint of 1026 psig

| | | |
|---|---|---|
| A. | 895 psig | INCORRECT: Plausible because the examinee could believe that the turbine bypass valves control at the setpoint prior to the reactor trip. Incorrect because the turbine bypass valves will be in hand and will not open at an automatic setpoint. |
| B. | 965 psig | INCORRECT: Plausible because this is the setpoint the turbine bypass valves control at during a reactor startup. The examinee could believe that since the reactor is tripped, that the turbine bypass setoint reverts to the startup setpoint. This is incorrect because the turbine bypass valves will not open in automatic when the ICS module is in HAND. |
| C. | 1010 psig | INCORRECT: Plausible because the examinee could believe that the normal trip setpoint is still in effect. This is incorrect because the turbine bypass valves will not open in automatic when the ICS module is in HAND. |
| D. | 1026 psig | CORRECT ANSWER: The atmospheric dump valves will open and control at their setpoint of 1026 psig. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 2 | | |
| | K/A # | 041 | A1.02 | |
| | Importance Rating | 3.1 | | 3.2 |

K/A: Steam Dump System (SDS)/Turbine Bypass Control: Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SDS controls including: Steam Pressure.

Proposed Question: RO Question #56

Technical Reference(s): TQ-TM-104-411-C001, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-7

Question Source: Bank #

Modified Bank # 880449

New

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question History: | Unmodified on System Exam 4 | Last NRC Exam: | N/A |

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 b.11

55.43

Comments:

KA Match: This question matches the KA because the examinee must know how the the Steam Dump System / Turbine Bypass Valves relate to each other to prevent exceeding their design pressure and lifting the Main Steam Safety Valves.

Plant conditions:

- Reactor at 100% power
- Turbine Bypass Valves MS-V-3D, MS-V-3E and MS-V-3F are closed in HAND for ICS module replacement

On a Reactor trip, OTSG ___(1)___ steam pressure will be ___(2)___ than the normal post-trip value.

A. (1) 1A
(2) lower

B. (1) 1A
(2) higher

C. (1) 1B
(2) lower

D. (1) 1B
(2) higher

Answer: B

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**57** **ID: 1720668** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Sequence of Events:

- The 'B' OTSG develops a steam leak in the Reactor Building.
- The crew manually trips the reactor based on rising Reactor Building pressure.
- A 4 PSIG ESAS actuation occurs.
- Current plant conditions:
  - RCS $T_{ave}$ is 560°F and lowering slowly.
  - OTSG 1A level is 86 inches on the Startup Range and slowly lowering.
  - OTSG 1B level is 86 inches on the Startup Range and slowly lowering.
  - OTSG 1A pressure is 985 psig and steady.
  - OTSG 1B pressure is 950 psig and lowering very slowly.
  - Main Feedwater flow to OTSG 1A is $0.3 \times 10^5$ lbm/hr.
  - Main Feedwater flow to OTSG 1B is $1.9 \times 10^5$ lbm/hr

The 'B' OTSG must be isolated in accordance with which procedure?

A. OP-TM-EOP-001, REACTOR TRIP

B. OP-TM-EOP-010, Guide 12, RCS STABILIZATION

C. OP-TM-AOP-051, SECONDARY SIDE HIGH ENERGY LEAK

D. OP-TM-EOP-003, EXCESSIVE PRIMARY TO SECONDARY HEAT TRANSFER

Answer: A

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) If a steam leak in the reactor building occurs, OP-TM-AOP-051 would be entered; (2)When the Reactor Building pressure approaches 2 psig (which could happen on an OTSG steam leak), the crew will trip the reactor; (3) To trip the reactor the crew will perform the EOP-001 Immediate Manual Actions and then perform a symptom check; (4) In this case, the crew would be sensitive to symptoms of excessive heat transfer, which in accordance with the OS -24 definition, does NOT exist at this time; (5) The operating crew would proceed with EOP-001 actions until they reached step 3.16 which would isolate the affected OTSG per Attachment 1.

| | | |
|---|---|---|
| A. | OP-TM-EOP-001, REACTOR TRIP | CORRECT: See above. |
| B. | OP-TM-EOP-010, Guide 12, RCS STABILIZATION | INCORRECT: Plausible because Guide 12 is implemented from Rule 3 (which is performed in EOP-003). There are steps in Guide 12 to isolate the OTSG from the condenser, but that is not applicable in this question. |
| C. | OP-TM-AOP-051, Secondary Side High Energy Leak. | INCORRECT: Plausible because the crew does enter this procedure, but incorrect because this procedure doesn't isolate the OTSG for a leak in the Reactor Building, but does direct a reactor trip at 2 psig in the Reactor Building. |
| D. | OP-TM-EOP-003, Excessive Primary to Secondary Heat Transfer. | INCORRECT: Plausible since EOP-003 would isolate the OTSG if symptoms of excessive heat transfer existed. Incorrect because those symptoms to not exist. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 103 2.4.6 | |
| | Importance Rating | 3.7 | 4.7 |

K/A: Containment System: Knowledge of EOP mitigation strategies.

Proposed Question: RO Question # 57

Technical Reference(s): OP-TM-EOP-001, Rev 16

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-001-PCO-4

Question Source: Bank # 908527

Modified Bank #

New

Question History: N/A Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.10 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must know the mitigation strategy for an OTSG with a steam leak in Containment.

High Cog: This question is high cog because the examinee must analyze the conditions in the stem to determine that the symptoms of excessive heat transfer do NOT exist.

**58** **ID: 1720680** **Points: 1.00**

Plant Conditions:
- 65% reactor power.
- ULD is in manual, all other ICS stations are in auto.
- Both SASS modules are de-energized.

EVENT:

- RC-P-1C breaker opens.

The output of RC-12-TAS Tave Auto/Manual Switch will __(1)__ because __(2)__.

A. (1) remain at the Loop A&B Average
(2) SASS is inoperable

B. (1) remain at the Loop A&B Average
(2) reactor power is below the ICS Runback setpoint

C. (1) swap to Loop A Average
(2) the RC-P-1C breaker is open

D. (1) swap to Loop A Average
(2) this loop has the highest RCS flow

Answer: D

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) RC-12-TAS Selector Switch is normally selected to Loop A&B Average; (2) The logic monitors RCS flow and if a RCS Loop Flow <9 6%, RC-12-TAS automatically swaps to the RCS Loop with the most flow; (3) which is in this case is the Loop A Average; (4) RCS Flow input to Tave selector switch is NOT an input to SASS, so the status of SASS has NO effect on the selector switch.

| | | |
|---|---|---|
| A. | (1) remain at the Loop A&B Average<br>(2) SASS is inoperable | INCORRECT: Plausible if the examinee thinks RCS Flow is a SASS input to the logic for RC-12-TAS. RCS flow is a parameter that is SASSED for input to ICS. |
| B. | (1) remain at the Loop A&B Average<br>(2) reactor power is below the ICS Runback setpoint | INCORRECT: Plausible because in most cases a RCP trip would cause a plant runback, but in this case the plant it below the runback setpoint. No runback would occur from a loss of this reactor coolant pump. If the examinee believes that since NO runback took place, that RC-12-TAS would not swap because reactor power is below the ICS runback setpoint. |
| C. | (1) swap to Loop A Average<br>(2) the RC-P-1C breaker is open | INCORRECT: Plausible since RC-12-TAS would swap to Loop A Average but NOT because because the RCP breaker is open. The correct input is RCS Flow. An RCP Breaker trip is input to the ICS runback circuit. |
| D. | (1) swap to Loop A Average<br>(2) this loop has the highest RCS flow | CORRECT: See above |

Examination Outline Cross-reference:

| | | |
|---|---|---|
| Level | RO | SRO |
| Tier # | 2 | |
| Group # | 1 | |
| K/A # | 003 | K5.03 |
| Importance Rating | 3.1 | 3.5 |

K/A: Reactor Coolant Pump System: Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP shutdown on T-ave., including the reason for the unreliability of T-ave. in the shutdown loop

Proposed Question: RO Question #58

Technical Reference(s): TQ-TM-104-624-C001, Rev 4 D554566 Analog Logic

Proposed References to be provided to applicants during examination: None

Learning Objective: 624-GLO-10

Question Source: Bank #

Modified Bank #

New X

| | | | |
|---|---|---|---|
| Question History: N/A | | Last NRC Exam: N/A | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.5 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must know the operational implication of a loss of flow in one reactor coolant loop, and the effect on the Tave system.

High Cog: This question is high cog because the examinee must analyze the conditions in the stem to understand the plant response for a reactor coolant pump trip.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**59** **ID: 1720699** **Points: 1.00**

With the Enable/Defeat switch for DC-V-2A/2B Decay Closed Cooler Inlet and DC-V-65A/65B Decay Closed Cooler Bypass placed in the DEFEAT position, this would result in DC-V-2A/2B __(1)__ AND DC-V-65A/65B __(2)__ .

A. (1) full closed
(2) full open

B. (1) full closed
(2) full closed

C. (1) full open
(2) full closed

D. (1) full open
(2) full open

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) There are key switches in the control room which need to be in the ENABLE position to be able to control each respective set of valves (DC-V-2A/65A and DC-V-2B/65B); (2) When the switches are in DEFEAT a 3 way solenoid valve fails each set of valves to their full cooling positions (DC-V-2A/2B full OPEN, and DC-V-65A/65B full CLOSED); (3) This ensures full cooling to the Decay Heat Removal System on an ES actuation.

| | | |
|---|---|---|
| A. | (1) full closed (2) full open | INCORRECT: Plausible if the examinee believes that the purpose of the key switch is to fail the valves to the NO COOLING position. |
| B. | (1) full closed (2) full closed | INCORRECT: Plausible if the examinee believes the purpose of the key switch is to start with the valves fully closed and have a more controlled transition to using the DHR system for cooling. A flow path would still exist through the various pumps that the Decay Closed system provides cooling for. |
| C. | (1) full open (2) full closed | CORRECT: See above. |
| D. | (1) full open (2) full open | INCORRECT: Plausible if the examinee believes that the purpose of the switch in defeat is to fail the valves in a position where the pumps would have a maximum flow. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 2 | | |
| | K/A # | 005 | A4.02 | |
| | Importance Rating | 3.4 | | 3.1 |

K/A: Residual Heat Removal System: Ability to manually operate and/or monitor in the control room: Heat exchanger bypass flow control

Proposed Question: RO Question #59

Technical Reference(s): TQ-TM-104-533-C001, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 533-GLO-4

Question Source: Bank # 468123

Modified Bank #

New

Question History: System Exam 10 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

| | | |
|---|---|---|
| | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.41 | b.7 |
| | 55.43 | |

Comments:

KA Match: This question matches the KA because the examinee must know how to operate the heat exchanger bypass flow valve from the control room.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**60** **ID: 1720769** **Points: 1.00**

Which one of the following Alternate Emergency Boration source tanks is required in accordance with 1301-1, SHIFT AND DAILY CHECKS.

Boric Acid Mix Tank (BAMT)
Borated Water Storage Tank (BWST)
Reactor Coolant Bleed Tanks (RCBTs)
Reclaimed Boric Acid Tanks (RBATs)

A. BWST OR RCBT

B. BAMT OR RCBT

C. BAMT OR RBAT "A" OR RBAT "B"

D. BWST OR RBAT "A" OR RBAT "B"

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) When the TMI-1 is in a condition greater than HOT SHUTDOWN, a reactor operator performs Shift and Daily checks to ensure technical specifications are being met on key equipment to ensure compliance with LCOs. (2) One of the checks is to check what the alternate emergency boration source is and to ensure the tank meets the requirements to the alternate source; (3) The procedure identifies the BAMT, RBAT "A", or RBAT "B" as those sources.

| | | |
|---|---|---|
| A. | BWST OR RCBT | INCORRECT: Plausible since the BWST is a source of emergency boration but is not considered an alternate source. In addition, two of the three RCBTs contain a boron concentration of equal to or greater than the current RCS boron, so it is plausible that they could be considered a source as well. |
| B. | BAMT OR RCBT | INCORRECT: Plausible since the BAMT is one source and two of the three RCBTs contain a boron concentration of equal to or greater than the current RCS boron, so it is plausible that they could be considered a source as well. |
| C. | BAMT OR RBAT "A" or RBAT "B" | CORRECT: See above. |
| D. | BWST OR RBAT "A" or RBAT "B" | INCORRECT: Plausible since the RBATs are an alternate source but not the BWST. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 024 | 2.2.12 |
| | Importance Rating | 3.7 | 4.1 |

K/A: Emergency Boration: Knowledge of surveillance procedures.

Proposed Question: RO Question # 60

Technical Reference(s): 1301-1, Rev 176 TQ-TM-104-561-C001, Rev 4

OP-TM-EOP-010, Guide 1, Rev 19

Proposed References to be provided to applicants during examination: NONE

Learning Objective: 561-GLO-7

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | b.10 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the procedure 1301-1, SHIFT AND DAILY CHECKS, is an surveillance procedure that checks the status of technical specification LCOs. Within that procedure, there is a section to ensure the alternate emergency boration status.

**61** **ID: 1720771** **Points: 1.00**

During normal 100% power operations, the steam supply to the Main Feedwater Pumps is the outlet of the __(1)__ , and shortly following a Reactor/Turbine trip, the source of steam to the Main Feedwater Pumps is __(2)__.

A. (1) "D" AND "F" Moisture Separators
(2) Main Steam

B. (1) "D" AND "F" Moisture Separators
(2) Auxiliary Steam

C. (1) HP Turbine prior to "C" AND "D" Moisture Separators
(2) Main Steam

D. (1) HP Turbine prior to "C" AND "D" Moisture Separators
(2) Auxiliary Steam

Answer: A

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) At 100% power, due to the design of the Main Feedwater Pump poppets, they prefer the lower pressure steam from the outlet of the "D" and "F" Moisture Separators; (2) This steam goes away when the Reactor/Main Turbine are shutdown (or tripped as in this question) so the Main Feedwater Pumps transition to the Main Steam System (via MS-V-5A/B) for their source of steam (approximately 5% to 24% power); (3) Auxiliary Steam is used during shutdown conditions.

| | | |
|---|---|---|
| A. | (1) "D" AND "F" Moisture Separators<br>(2) Main Steam | CORRECT: See above |
| B. | (1) "D" AND "F" Moisture Separators<br>(2) Auxiliary Steam | INCORRECT: Plausible because Auxiliary Steam is used as a source of steam for the Main Feedwater Pumps. Incorrect because it isn't used until the Auxiliary Boilers are running after a reactor trip. |
| C. | (1) HP Turbine prior to "C" AND "D" Moisture Separators<br>(2) Main Steam | INCORRECT: Plausible because there are steam line connections at that location. Incorrect because those steam lines to go feedwater heating. |
| D. | (1) HP Turbine prior to "C" AND "D" Moisture Separators<br>(2) Auxiliary Steam | INCORRECT: Plausible because there are steam line connections at that location. Incorrect because those steam lines to go feedwater heating. In addition Auxiliary Steam is not used right after a reactor trip. |

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| Level | RO | | SRO |
| Tier # | 2 | | |
| Group # | 1 | | |
| K/A # | 039 | K1.08 | |
| Importance Rating | 2.7 | | 2.9 |

K/A: Main and Reheat Steam System: Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: MFW

Proposed Question: Question #61

Technical Reference(s): 302-011 | 302-041
TQ-TM-104-411-C001, Rev 8 | TQ-TM-104-431-C001, Rev 4

Proposed References to be provided to applicants during examination: None

Learning Objective: 411-GLO-5

Question Source: Bank #
Modified Bank # 355620
New

Question History: N/A Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must know the physical connection between the Main Steam System and the Main Feedwater Pumps, in addition the cause-effect of a reactor trip on both systems.

Five minutes after a Reactor/Turbine Trip from 100% power, the source of steam to the main feedwater pumps is the:

A. Main Steam System.

B. Auxiliary Steam System.

C. Outlet of the "C" Moisture Separator.

D. 4th stage of the Extraction Steam System.

Answer: A

# EXAMINATION ANSWER KEY
16-01 SENIOR REACTOR OPERATOR NRC EXAM

**62** **ID: 1720783** **Points: 1.00**

Plant Condition:

- 100% power with ICS in full auto.

EVENT

- Loss of Offsite Power

Subsequently:

- All 8 (eight) HSPS switches placed in DEFEAT.
- No other operator actions have been taken.

What is the setpoint from HSPS for OTSG level control?

A. 0" on the startup range.

B. 25" on the startup range.

C. 50% on the operating range.

D. 75-85% on the operating range.

Answer: C

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) On a Loss of Offsite Power, all Reactor Coolant Pumps trip; (2) When this happens, the HSPS setpoint sent to the EF-V-30s for OTSG level control is 50% in the operating range to promote natural circulation; (3) When the HSPS switches are taken to DEFEAT, the 50% operating range signal is still locked in, until the operator breaks the 'seal-in' relay by taking each EF-V-30 to manual; (4) Since the stem does not indicate that this happens, the setpoint remains 50% in the operating range.

| | | |
|---|---|---|
| A. | 0" on the startup range. | INCORRECT: Plausible because the examinee could believe that HSPS has lost power on the loss of offsite power, and that OTSG level control will have to be in manual. Incorrect because HSPS is powered from the Vital Busses, and they do not lose power on a loss of offsite power. |
| B. | 25" on the startup range. | INCORRECT: Plausible because on a loss of offsite power, both Main Feedwater Pumps would trip. When the Main Feedwater Pumps trip, HSPS sends a signal to control OTSG level at 25" in the startup range. Incorrect because the loss of reactor coolant pump signal has priority. |
| C. | 50% on the operating range | CORRECT: See above |
| D. | 75-85% on the operating range. | INCORRECT: Plausible because the operator would feed to this range in accordance with Rule 4 of OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS if subcooling margin were gone. Feeding this high would also promote natural circulation more. Incorrect because the operator has to take manual control to feed to this level. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 2 | | |
| | K/A # | 061 | A3.03 | |
| | Importance Rating | 3.9 | | 3.9 |

K/A: Auxiliary/Emergency Feedwater (AFW) System: Ability to monitor automatic operation of the AFW, including: AFW S/G level control on automatic start

Proposed Question: RO Question #62

Technical Reference(s): TQ-TM-104-644-C001, Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: 644-GLO-5

Question Source: Bank #

Modified Bank # 719815

New

Question History: System Exam 9 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 b.7

55.43

Comments:

KA Match: This question matches the KA because the examinee must know the automatic operation of the EFW system that should occur on a loss of offsite power, which includes the signal that HSPS controls OTSG level.

High Cog: This question is high cog because the examinee must know the plant response on a loss of offsite power, and the effect on HSPS level control.

Plant Conditions:

- Loss of BOTH FW pumps
- Reactor and Turbine trip

Event:

- All 8 (eight) HSPS switches placed in DEFEAT.
- No other operator actions have been taken.

What is the setpoint, from HSPS, for OTSG level control?

A. 0" on the startup range.

B. 25" on the startup range.

C. 50% on the operating range.

D. 75-85% on the operating range.

Answer: B

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**63** **ID: 1724247** **Points: 1.00**

Plant Conditions:

- 40% power with ICS in auto.
- Main Condenser Tube Leak on the "A" side.
- Circulating Water Pumps, CW-P-1A, CW-P-1B, CW-P-1C are shutdown.
- Loop Cross Connect Valves, CW-V-4A/4B are closed.

EVENT:

- CW-P-1D Trips.
- Main Condenser vacuum starts to lower.
- CRS directs the CRO to trip the Main Turbine.
- Main Condenser vacuum lowers to 25" Hg and stabilizes.

Based on the conditions, the reactor has __(1)__ and OTSG pressure control is being controlled by the __(2)__.

A. (1) tripped
(2) Turbine Bypass Valves

B. (1) tripped
(2) Atmospheric Dump Valves

C. (1) stabilized at ~18% power
(2) Turbine Bypass Valves

D. (1) stabilized at ~18% power
(2) Atmospheric Dump Valves

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) OTSG pressure control transfers from the Turbine Bypass Valves (TBVs) to the Atmospheric Dump Valves (ADVs) when there are < 2 Circulating Water Pumps operating or when Main Condenser Vacuum is < 23" Hg; (2) In this question, when CW-P-1D trips, this still leaves 2 Circulating Water Pumps operating, and since Main Condenser Vacuum never lowers to less than 23" Hg, the TBVs retain OTSG pressure control; (3) Due to reactor power being below 45%, when the Main Turbine is tripped, the reactor will not automaticaly trip, but will runback to approximately 18% power.

| | | |
|---|---|---|
| A. | (1) tripped<br>(2) Turbine Bypass Valves | INCORRECT: Plausible it the examinee does not recognize that at 40% power the Reactor will not trip on a Turbine Trip. Incorrect because the reactor does not trip. |
| B. | (1) tripped<br>(2) Atmospheric Dump Valves | INCORRECT: Plausible it the examinee does not recognize that at 40% power the Reactor will not trip on a Turbine Trip. Incorrect because the reactor will not trip,and the ADVs will not have OTSG pressure control. |
| C. | (1) stabilized at ~18% power<br>(2) Turbine Bypass Valves | CORRECT: See above |
| D. | (1) stabilized at ~18% power<br>(2) Atmospheric Dump Valves | INCORRECT: Plausible because the examinee could believe that the ADVs have OTSG pressure control. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 075 | 2.4.21 |
| | Importance Rating | 4.0 | 4.6 |

K/A: Circulating Water System: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.

Proposed Question: RO Question # 63

Technical Reference(s): ICS Digital Print D553854; TQ-TM-104-621-C001, Rev 8; TQ-TM-104-641-C001, Rev 2; OP-TM-411-000, Rev 21

Proposed References to be provided to applicants during examination: None

Learning Objective: 641-GLO-10

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee have knowledge relative to the OTSG pressure control logic with respect to operation of the CW Pumps. In-addition requires knowledge related to heat production because of the runback and heat removal.

High Cog: This question is high cog because the examinee analyzes plant conditions and has to integrate the relationship between the operation of several systems.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**64** **ID: 1736357** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Electrical load reducing, with a constant reactor power.
- OTSG 1A steam pressure is 870 psig and decreasing.
- RB pressure is 0.3 psig and steady.
- Total FW flow is 10.2 E6 lbm/hr.
- PLF-1-9 and PLF-1-10 for Intermediate Bldg Fire AND Intermediate Bldg Trouble
- AH-E-73 (Intermediate Bldg.) discharge high temperature alarm.
- PPC Point A0331 TURB DRIV EF-P1 BRG COOL WTR OUT is in HI-2 Alarm
- RM-A-2G reading normal.

What action is required for the above event?

A. Dispatch personnel to look for the steam leak.

B. Manually trip the Reactor AND go to OP-TM-EOP-001.

C. Reduce power to < 45% and manually trip the Main Turbine.

D. Initiate a Plant Shutdown IAW 1102-4, POWER OPERATION.

Answer: B

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) Based on the symptoms there is steam leak in the Intermediated Building; (2) OP-TM-AOP-051, SECONDARY SIDE HIGH ENERGY LEAK, Section 5, directs if there is a leak in the Intermediate Building with a Hi-2 alarm on safety related equipment to trip the Reactor and GO TO OP-TM-EOP-001, REACTOR TRIP; (3) EF-P-1 is safety related equipment.

| | | |
|---|---|---|
| A. | Dispatch personnel to look for the steam leak. | INCORRECT: Plausible if the examinee is not sure of leak location. The procedure does address looking for the leak when it is in the Reactor Building. |
| B. | Manually trip the Reactor AND go to OP-TM-EOP-001 | CORRECT: See above. |
| C. | Reduce power to < 45% and manually trip the Main Turbine | INCORRECT: Plausible if the examine believes that since the steam leak is in the Intermediate Building they could initiate a plant shutdown and when below 45%, then the turbine could be tripped without an automatic reactor trip. |
| D. | .Initiate a Plant Shutdown IAW 1102-4, POWER OPERATION. | INCORRECT: Plausible since this step appears several time the in OP-TM-AOP-051, however the HI-2 Alarm would require tripping the Reactor. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 1 | |
| Group # | 1 | |
| K/A # | 040 | AA2.02 |
| Importance Rating | 4.6 | 4.7 |

K/A: Steam Line Rupture: Ability to determine and interpret the following as they apply to the Steam Line Rupture: Conditions requiring a reactor trip

Proposed Question: RO Question #64

Technical Reference(s): OP-TM-AOP-051, Rev 2

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-051-PCO-5

Question Source: Bank #

Modified Bank # 462996

New

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

| | | |
|---|---|---|
| | Comprehension or Analysis | X |
| 10 CFR Part 55 Content: | 55.41 b.5 | |
| | 55.43 | |

Comments:

KA Match: This question matches the KA because the examinee must know the that the reactor must be tripped with a steam line rupture in the intermediate building if there is a HI-2 alarm on safety related equipment.

High Cog: This question require analysis of plant conditions and based on these conditions, the priority of required actions.

The plant has been operating at full power when the following indications are received:

- Electrical load reducing, with a constant reactor power.
- OTSG 1A steam pressure is 870 psig and decreasing.
- RB pressure is 0.3 psig and steady.
- Total FW flow is 10.2 E6 lbm/hr.
- AH-E-73 (Intermediate Bldg.) discharge high temperature alarm.
- PPC Point A0331 TURB DRIV EF-P1 BRG COOL WTR OUT is in alarm and currently at 123 F and rising at 2 degrees every minute
- AH-E-24A (EFW pump area) trips on high discharge temperature.
- RM-A-2G reading normal.

What action is required for the above event?

A. Manually trip the Reactor AND go to OP-TM-EOP-001.

B. Reduce power to < 45% and manually trip the Main Turbine.

C. Reduce power at a rate specified by the CRS until an RPS trip setpoint is reached.

D. Verify no personnel are in the Intermediate Building AND continue Power Operations.

Answer: A

**65** **ID: 1718268** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- Building Spray Pump, BS-P-1B, is out of service.

SEQUENCE OF EVENTS:

1. LOCA
2. RCS pressure lowers to 890 psig.
3. Reactor Building Pressure rises to 32 psig.
4. 1S 480V ES Bus trips.

Post Accident Reactor Building cooling is ___(1)___, because ___(2)___.

A. (1) adequate
(2) one Building Spray Pump and one Reactor Building Emergency Cooling Fan (AH-E-1) are operating

B. (1) adequate
(2) one Building Spray Pump and two Reactor Building Emergency Cooling Fans (AH-E-1) are operating

C. (1) NOT adequate
(2) only one Building Spray Pump and one Reactor Building Emergency Cooling Fan (AH-E-1) are operating

D. (1) NOT adequate
(2) only one Building Spray Pump and two Reactor Building Emergence Cooling Fans (AH-E-1) are operating

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) The post-accident reactor building emergency cooling is adequate with one Building Spray Pump and one Reactor Building Emergency Cooling Fan; (2) In this question, the examinee is starting with one Building Spray Pump out of service, the other Building Spray pump remains operable; (3) When the ES occurs (indicated by RCS pressure lowering to 890 psig), the transfer of 1C ES Valves 480V bus is inhibited and will remain powered from 1S 480V ES Bus; (4) When 1S 480V ES bus trips, AH-E-1B and AH-E-1C lose power, thus leaving only AH-E-1A operable.

| | | |
|---|---|---|
| A. | (1) adequate<br>(2) one Building Spray Pump and one Reactor Building Emergency Cooling Fan (AH-E-1) is operating | CORRECT: Correct Answer |
| B. | (1) adequate<br>(2) one Building Spray Pump and two Reactor Building Emergency Cooling Fans (AH-E-1) are operating | INCORRECT: Plausible because the examinee could believe that 1C ES Valves MCC swapped to 1P 480V ES Bus, which would have powered AH-E-1C. Incorrect because the ES blocked 1C ES Valves from swapping to 1P 480V ES bus, thus leaving AH-E-1C without power. |
| C. | (1) NOT adequate<br>(2) only one Building Spray Pump and one Reactor Building Emergency Cooling Fan (AH-E-1) is operating | INCORRECT: Plausible because the examinee could believe that having only one Building Spray pump and one RB Emergency Cooling fan operating is inadequate post accident cooling. Incorrect because having one of each running is adequate. |
| D. | (1) NOT adequate<br>(2) only one Building Spray Pump and two Reactor Building Emergence Cooling Fans (AH-E-1) are operating | INCORRECT: Plausible because the examinee could believe that having only one Building Spray pump and two RB Emergency Cooling fans is inadequate post accident cooling. Incorrect because only one of each running is adequate. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # 2 | 2 | |
| Group # 1 | 1 | |
| K/A # | 026 | K3.01 |
| Importance Rating | 3.9 | 4.1 |

K/A: Containment Spray System: Knowledge of the effect that a loss or malfunction of the CSS will have on the following: CCS

Proposed Question: RO Question #65

Technical Reference(s): TQ-TM-104-642-C001, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: 642-GLO-10

Question Source: Bank #

| | | | |
|---|---|---|---|
| | Modified Bank # | | |
| | New | X | |
| Question History: | N/A | Last NRC Exam: | NO |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must know the effect that a loss of one Building Spray Pump and two RB Emergency Cooling fans will have on the containment cooling system.

High Cog: This question is High Cog because the examinee must analyze the stem and recognize that an ES signal has occurred and how the loss of power has affected the Building Spray and RB Emergency Cooling Systems.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**66** **ID: 1736429** **Points: 1.00**

Plant Conditions:

- 90% power with ICS in auto.

EVENT:

- PLB-1-7 INSTRUMENT AIR PRESS LOW TURBINE AREA comes in.
- PI-1403 Secondary IA pressure reads 59 psig.
- PI-222 Primary Instrument Air pressure reads 90 psig.

Based on the above plant conditions, the initial action the operating crew must perform is:

A. Trip the Reactor.

B. Initiate a plant shutdown.

C. Dispatch a NLO to isolate Primary IA from Secondary IA.

D. Start Instrument Air Compressors, IA-P-1A and IA-P-1B from the Control Room.

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1)OP-TM-AOP-028, LOSS OF INSTRUMENT AIR is entered when pressure on PI-1403 < 80 psig; (2) When pressure on PI-1403 goes less than 60 psig, the next steps are to dispatch operators to block open MU-V-20 and IC-V-3 and 4, then trip the reactor. (3) Of the actions listed, the operating crew would trip the reactor.

| | | |
|---|---|---|
| A. | Trip the Reactor. | INCORRECT: See above |
| B. | Initiate a plant shutdown. | INCORRECT: Plausible if the examinee believes that due to the loss of instrument air a Technical Specification LCO must be entered, and a plant shutdown must occur. Incorrect the reactor must be tripped when PI-1403 reads less than 60 psig.Plausible since this would be done if Primary IA>80- and Secondary < 60 AND the Reactor was not tripped per procedure requirements. Then the procedure directs to restore Secondary Pressure. |
| C. | Dispatch a NLO to isolate Primary IA from Secondary IA. | INCORRECT: INCORRECT: Plausible since this could raise instrument air pressure. Incorrect because the operating crew must tripped when PI-1403 < 60 psig. |
| D. | Start Instrument Air Compressors, IA-P-1A and IA-P-1B from the Control Room. | INCORRECT: Plausible because this is an action in OP-TM-AOP-028 to take if IA-PI-491 < 85 psig. Incorrect because the reactor must be tripped when PI-1403 is less than 60 psig. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 065 | AA2.06 |
| | Importance Rating | 3.6 | 4.2 |

K/A: Loss of Instrument Air: Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to trip reactor if instrument air pressure is de-creasing

Proposed Question: RO Question # 66

Technical Reference(s): OP-TM-AOP-028, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: 850-PCO-5

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | b.7 | |

Comments:

KA MATCH: This question matches the KA because the examinee must know when the reactor must be tripped on a loss of instrument air.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**67** **ID: 1736434** **Points: 1.00**

Plant Conditions:

- The Plant has been shutdown for 24 days.
- Reactor Vessel level is being lowered in support of internal Reactor Vessel work.
- The "B" Train of Decay Heat Removal (DHR) is operating with a flow rate of 1,400 gpm.

EVENT:

- PLB-5-5 1B DECAY HEAT REMOVAL COMPARTMENT LEAK DETECT comes in.
- Decay Heat Pump, DH-P-1B, suction line temperature indicator has risen 10.5°F.
- RCS Cold Leg level is 15 inches above centerline and slowly lowering.
- DH-P-1B motor amperage and flow is significantly oscillating.

Based on the above conditions, which ONE of the following actions must be taken FIRST?

A. Start "A" Train of Decay Heat Removal.

B. Lower flowrate from DH-P-1B to less than 1,200 gpm.

C. Immediately place DH-P-1B in the "Pull-to-Lock" position.

D. Raise RCS level until Decay Heat removal Pump operating indications are normal.

Answer: C

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) The OP-TM-EOP-030, LOSS OF DECAY HEAT REMOVAL, entry criteria are met: Incore temperature increases > 10F due to an unplanned condition; (2) In addition, since flow and motor amperage are oscillating, in accordance with Step 2.1, DH-P-1B must be placed in PTL.

| | | |
|---|---|---|
| A. | Start "A" Train of Decay Heat Removal. | INCORRECT: Plausible since the follow up actions direct this action, however the priority is to place DH-P-1B in PTL. In-addition, starting the alternate pump could cause a common mode failure with lowering level. |
| B. | Lower flowrate from DH-P-1B to less than 1,200 gpm. | INCORRECT: Plausible since lowering flowrate from DH-P-1B to less than 1,200 gpm would place parameters into a more acceptable region. This flow rate would still not be allowed. |
| C. | Immediately place DH-P-1B in the "pull-to-lock" position. | CORRECT: See above |
| D. | Raise RCS level until Decay Heat removal Pump operating indications are normal. | INCORRECT: Plausible since Raising RCS level until Decay Heat removal Pump operating indications are normal would maintain level in the acceptable region and possibly reduce oscillating pump parameters, but not allowed under these conditions, since EOP-030 requires stopping the pump given the initial conditions. |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 1 | | |
| | Group # | 1 | | |
| | K/A # | 025 | AA2.01 | |
| | Importance Rating | 2.7 | | 2.9 |

K/A: Loss of Decay Heat Removal: Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Proper amperage of running LPI/decay heat removal/RHR pump(s)

Proposed Question: RO Question # 67

Technical Reference(s): OP-TM-EOP-030, Rev 9

OP-TM-212-000, Rev 23

Proposed References to be provided to applicants during examination: OP-TM-212-000, Attachment 7.2 Min Height of Water to Avoid Vortex Formation vs DH Flow

Learning Objective: EOP-030-PCO-4

Question Source: Bank #

Modified Bank # 304375

# EXAMINATION ANSWER KEY



New

Question History: Sim Exam 9 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

KA MATCH: This question matches the KA because the examinee must interpret the oscillating running amperage of a DHR pump to determine that it is cavitating.

High Cog: The examinee must analyze the stem to determine the entry criteria for OP-TM-EOP-030 are met, and then determine that DH-P-1A must be place in PTL. In addition, the examinee can check the answer to determine that the pump is below the vortex limit.

Plant Conditions:

- The Plant has been Shutdown for 24 days.
- Reactor Vessel level is being lowered in support of internal Reactor Vessel work.
- The "B" Train of Decay Heat Removal (DHR) is operating with a flow rate of 1,400 gpm.
- The "A" Train of DHR is out-of-service for minor maintenance and is expected to be returned to service within the next 4 hours.

Event:

- Decay Heat Pump, DH-P-1B, suction line temperature indicator has risen 10.5°F.
- RCS Cold Leg level is 10.5 inches above centerline.
- DH-P-1B Motor Amperage is significantly oscillating.

Based on the above conditions, which ONE of the following actions will be taken FIRST?

A. Lower flowrate from DH-P-1B to less than 1,200 gpm.

B. Immediately place DH-P-1B in the "pull-to-lock" position.

C. Raise RCS level to at least 14 inches above Hot Leg centerline.

D. Actuate the RB Evacuation alarm and evacuate ALL personnel from the RB.

Answer: B

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**68** **ID: 1736443** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- MU-P-1A (Make-up Pump) is supplying normal make-up and Seal Injection.
- MU-P-1B is in Normal-After Stop.

SEQUENCE OF EVENTS:

- Loss of Coolant Accident.
- RCS pressure lowers to 1500 psig.
- RB pressure is 5 psig.
- Loss of Offsite Power.
- MU-P-1A Trips.

Based on the above conditions, if MU-P-1B is ES Selected using the 43 Selector Switch on the 1E 4160V Bus, MU-P-1B would __(1)__ and MU-P-1C would __(2)__.

A. (1) start and immediately trip, (2) trip.

B. (1) start and immediately trip, (2) continue to run.

C. (1) start and continue to run, (2) continue to run.

D. (1) start and continue to run, (2) trip.

Answer: D

**Answer Explanation**

RO EXAM CHOICE ORDER

A. (1) Start and Continue to run, (2) trip

B. (1) Start and immediately trip, (2) trip

C. (1) Start and continue to run (2) Continue to run.

D. (1) Start and immedeately trip, (2) continue to run.

Correct Answer: A

<<Explanation: To answer this question correctly, the examinee must know: (1) The LOCA occurs which causes an ES Actuation, both 1600 psig and 4 psig, this sends a start signal to MU-P-1A and MU-P-1C; (2) These are Block 1 ES loads, when the LOOP occurs, MU-P-1A and MU-P-1C stop and restart when the Emergency Diesel Generators start and load on the 1D and 1E 4160V busses; (3) When the 43 selector switch on the 1E 4160V Bus is rotated to ES select MU-P-1B, then MU-P-1B will start and MU-P-1C will trip.

| | | |
|---|---|---|
| A. | (1) start and continue to run, (2) trip. | CORRECT: See above. |
| B. | (1) start and immediately trip, (2) trip. | INCORRECT: Plausible if the examinee thinks that since MU-P-1B is green flagged that anti-pump would actuate when the pump is started and trip the pump. Incorrect because anti-pump would only actuate if a simultaneous trip and start signal. |
| C. | (1) start and continue to run, (2) continue to run. | INCORRECT: Part 1 is correct. Part 2 is plausible if the examinee does not understand the operation of the 43 selector switch and its effect on the pump trip circuit. This interlock exists to prohibit from running two Makeup Pumps on a Emergency Diesel Generator. |
| D. | (1) start and immediately trip, (2) continue to run. | INCORRECT: Plausible if the examinee thinks that since MU-P-1B is green flagged that anti-pump would actuate when the pump is started and trip the pump. Incorrect because anti-pump would only actuate if a simultaneous trip and start signal. Part 2 is plausible if the examinee does not understand the operation of the 43 selector switch and its effluent on the pump trip circuit. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 006 | K2.01 |
| | Importance Rating | 3.6 | 3.9 |

K/A: Emergency Core Cooling System: Knowledge of bus power supplies to the following: ECCS Pumps

Proposed Question: RO Question # 68

Technical Reference(s): 1105-3, Rev 53

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO-10

Question Source: Bank #

Modified Bank #

New X

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question History: N/A | | Last NRC Exam: N/A | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.7 | |
| | 55.43 | | |

Comments:
KA Match: This question matches the KA because the examinee must have knowledge of the power supply to the Makeup Pumps and knowledge of the interlock preventing from running two Makeup Pumps on an Emergency Diesel Generator.

High Cog: This question is High Cog because the examinee must analyze the plant conditions to determine the status of MU-P-1B after a LOOP and ES actuation.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**69** **ID: 1736454** **Points: 1.00**

Plant Conditions:

- 50% power with ICS in auto.

EVENT:

- MAP G-2-1 CRD PATTERN ASYMMETRIC alarms.
- Red Fault Indication lit on the Position Indication Panel (PIP).
- The Asymmetric Control Rod Fault indication is currently lit on the Diamond Control Panel.

These indications would be caused by a deviation of an individual rod position of __(1)__from the group average, which __(2)__, and would require entry into OP-TM-AOP-062, INOPERABLE ROD.

A. (1) 7 inches
(2) excludes the faulted rod

B. (1) 7 inches
(2) includes the faulted rod

C. (1) 9 inches
(2) excludes the faulted rod

D. (1) 9 inches
(2) includes the faulted rod

Answer: C

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) Each rod in a group is compared with the group average; (2) If a rod in a group deviates by more than 7 inches (5%) from the group average, an individual amber fault light for that rod will light on the PI panel; (3) A rod that is 9 inches (6.5%) from its group average, the faulted rod is removed from the group average calculation; (4) An individual red fault light for that rod will light on the PIP and the asymmetric fault light on the Operator Diamond Control Panel will be lit; (5) Both the 7" and 9" Asymmetric fault initiate alarm MAP G-2-1. (6) The 7" Asymmetric Rod does NOT require entry into AOP-062.

| | | |
|---|---|---|
| A. | (1) 7 inches<br>(2) excluding the Faulted Rod. | INCORRECT: Plausible since this initiates the G-2-1 Alarm. Incorrect because the examinee may think that the 7" also excludes the faulted rod as does the 9' fault. In addition, a 7" rod does not require entry into OP-TM-AOP-062. |
| B. | (1) 7 inches<br>(2) including the Faulted Rod. | INCORRECT: Plausible since this initiates the G-2-1 Alarm and the 7" fault does include the faulted rod. Incorrect because a 7" misaligned rod does not require entry into OP-TM-AOP-062. |
| C. | (1) 9 inches<br>(2) excluding the Faulted Rod. | CORRECT: See above |
| D. | (1) 9 inches<br>(2) including the Faulted Rod. | INCORRECT" Plausible since the 9" fault would cause all the above indications, requires entry into OP-TM-AOP-062, however does not include the faulted rod in the calculation. |

Examination Outline Cross-reference:

| | | | |
|---|---|---|---|
| Level | RO | | SRO |
| Tier # | 2 | | |
| Group # | 2 | | |
| K/A # | 014 | 2.4.4 | |
| Importance Rating | 4.5 | | 4.7 |

K/A: Rod Position Indication: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: RO Question # 69

Technical Reference(s): OP-TM-AOP-062, Rev 007 TQ-TM-104-622-C001, Rev 007

OP-TM-622-000, Rev 007

Proposed References to be provided to applicants during examination: None

Learning Objective: 622-GLO-10

Question Source: Bank #

Modified Bank # 782160

New

Question History: System Exam 14 Last NRC Exam: N/A

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.10 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must recognize the entry conditions to OP-TM-AOP-062 based on the misaligned rod indications.

High Cog: This question is high cog because the examinee must analyze the indications in the stem to determine if an AOP must be entered.

The Asymmetric control rod fault indication is currently lit on the Diamond control station.

Select the ONE choice below that correctly describes the indications that constitute an asymmetric control rod fault indication on the Diamond control station.

A. Deviation of an individual rod's position indication of 9 inches, (6.5%), from the average position of the remaining 60 full length control rods.

B. Deviation of an individual rod's position indication of 7 inches, (5%), from the average position indication of all of the control rods in that group. Faulted rod remains in the group average.

C. Deviation of an individual rod's position indication of 9 inches, (6.5%), from the average position indication of the other control rods in that group. Faulted rod remains in the group average.

D. Deviation of an individual rod's position indication of 9 inches, (6.5%), from the average position indication of all of the control rods in that group. Faulted rod is removed from the group average.

Answer: D

**70** **ID: 1736470** **Points: 1.00**

What is minimum personnel that must be in the Control Room in accordance with OP-TM-112-101-1002, SHIFT STAFFING REQUIREMENTS, when the RCS temperature is greater than 200F?

A. One SRO or one RO

B. One SRO AND one RO

C. One SRO AND two ROs

D. One SRO OR one RO AND one STA

Answer: B

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) OP-TM-112-101-1002 requires that a minimum of 1 SRO and 1 RO be in the Control Room at all times when RCS temperature is > 200F.

| | | |
|---|---|---|
| A. | One SRO or one RO | INCORRECT: Plausible since if RCS temperature were less than 200F this combination would be acceptable. |
| B. | One SRO AND one RO | CORRECT: See Above |
| C. | One SRO AND two ROs | INCORRECT: Plausible if the examinee interprets the table in step 4.1 to mean that the two RO's must be in the control room at all times. Incorrect because the table is referring to minimum shift staffing. |
| D. | One SRO OR one RO AND one STA | INCORRECT: Plausible if the examinee interprets the table in step 4.1 to mean that the one RO and the STA must be in the control room at all times. Incorrect because the table is referring to minimum shift staffing. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 1 | |
| | K/A # | 2.1.1 | |
| | Importance Rating | 3.8 | 4.2 |

K/A: Knowledge of conduct of operations requirements.

Proposed Question: RO Question #70

Technical Reference(s): OP-TM-112-101-1002, Rev 10 Tech Spec 6.2.2.2

Proposed References to be provided to applicants during examination: None

Learning Objective: Prewatch DBIG - APCO-1

Question Source: Bank # 372961

Modified Bank #

New

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 b.10

55.43

Comments: (KA Match, why high cog, why SRO only)

KA MATCH: This question matches the KA because the examinee must know the minimum shift staffing in the control room.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**71** **ID: 1736534** **Points: 1.00**

In accordance with Tech Specs, the Minimum Conditions for Criticality (Excluding Low Power Physics Testing) requires the RCS Temperature to be __(1)__ and the Safety Rod Groups to be __(2)__ prior to any other reduction in shutdown margin.

A. (1) ≥ 532F (2) fully withdrawn

B. (1) ≥ 532F (2) fully inserted

C. (1) >525F (2) fully withdrawn

D. (1) >525F (2) fully inserted

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) Tech Spec 3.1.3- The reactor coolant temperature shall be above 525°F except for port ons of low power physics test ng. (2) Safety rod groups shall be fully withdrawn prior to any other reduct on in shutdown margin by deborat on or regulat ng rod withdrawal during the approach to crit cality

| | | |
|---|---|---|
| A. | (1) ≥ 532F (2) fully withdrawn | INCORRECT: Plausible since the calculation for Estimated Critical Rod Position assumes RCS Temperature is 532F. |
| B. | (1) ≥ 532F (2) fully inserted | INCORRECT: Plausible since the calculation for Estimated Critical Rod Position assumes RCS Temperature is 532F and if the examinee thinks the Safety Rods must be inserted to maintain the Tech Spec required 1% SDM prior to withdrawing rods with the intent to go critical. |
| C. | (1) >525F (2) fully withdrawn. | CORRECT: See above |
| D. | (1) >525F (2) fully inserted | INCORRECT: Plausible if the examinee thinks the Safety Rods must be inserted to maintain the Tech Spec required 1% SDM prior to withdrawing rods with the intent to go critical. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| | Group # | 2 | |
| | K/A # | 2.2.38 | |
| | Importance Rating | 3.6 | 4.5 |

K/A: Equipment Control: Knowledge of conditions and limitations in the facility license.

Proposed Question: RO Question # 71

Technical Reference(s): 1103-8, Rev 55 T.S. 3.1.3, AMD 278

Proposed References to be provided to applicants during examination: None

Learning Objective: Normal Ops - APCO-1

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

| | |
|---|---|
| 10 CFR Part 55 Content: | 55.41 |
| | 55.43 |

Comments: (KA Match, why high cog, why SRO only)

KA Match: This question matches the KA because the examinee must have knowledge of the requirements to pull to criticality.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**72** **ID: 1737637** **Points: 1.00**

A surveillance for an ES pump is in progress.
The analog suction pressure gauge, normally isolated except for this surveillance, is broken.
The suction pressure is used to calculate pump head, which is a required parameter for the surveillance test.
An alternate instrument is available, but it is not listed in the surveillance procedure.

In accordance with ER-TM-321-1041, TMI-1 IST PROGRAM REQUIREMENTS, which one of the following identifies the required actions with respect to this surveillance?

A. Continue the surveillance. Annotate in the procedure that the alternate instrument is used.

B. Stop the surveillance. Evaluate if the alternate gauge can be used.

C. Stop the surveillance. The surveillance may only be run with the installed gauge.

D. Stop the surveillance. Perform an IC to allow use of the alternate instrument, then continue the surveillance.

Answer: B

**Answer Explanation**

<<<<Explanation: To answer this question correctly, the examinee must know: (1) ER-TM-321-1041, Attachment 1 addresses instrument requirements for an In-Service Test (IST); (2) If an instrument fails during and IST the correct actions are to : Stop the test, place the system in a safe configuration, locate a backup instrument or replace the existing instrument using Section 2 of the attachment, and resume or re-perform the test with the replaced instrument.

| | | |
|---|---|---|
| A. | Continue the surveillance. Annontate in the procedure that the alternate instrument is used. | INCORRECT: This is incorrect, ER-TM-321-1041 directs the surveillance to be stopped, and then evaluate if the new instrument is adequate to perform the surveillance |
| B. | Stop the surveillance. Evaluate if the alternate gauge can be used. | . Correct Answer: See above. |
| C. | Stop the surveillance. The surveillance may only be run with the installed gauge. | Incorrect Answer: Although fixing the gauge and re-running the surveillance is an option, it is not the only option. The operators should evaluate if the alternate gauge meets the requirements to run the surveillance. |
| D. | Stop the surveillance. Perform an IC to allow the use of the alternate instrument, then continue the surveillance. | INCORRECT: Plausible because an IC would allow for the change of a procedure. Incorrect because this is not required to be performed, nor is it the correct administrative procedure to use when an instrument malfunction has occured during a surveillance. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 3 | |
| Group # | 2 | |
| K/A # | 2.2.12 | |
| Importance Rating | 3.7 | 4.1 |

K/A: Knowledge of surveillance procedures.

Proposed Question: Question 72

Technical Reference(s): ER-TM-321-1041, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP DBIG, APCO-2

Question Source: Bank # 504102

Modified Bank #

New

| | | | |
|---|---|---|---|
| Question History: None | | Last NRC Exam: None | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | b.10 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of the IST program requirements at TMI.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**73** **ID: 1736792** **Points: 1.00**

Plant Conditions:

- OP-TM-EOP-005, OTSG TUBE LEAKAGE has been entered for a 'A' OTSG tube rupture.
- Reactor is tripped.

'A' OTSG must be isolated when ___(1)___ AND ___(2)___.

A. (1) RCS pressure is less than 1000 psig,
(2) BWST level < 22 feet

B. (1) OTSG pressure is less than 1000 psig,
(2) BWST level < 22 feet

C. (1) RCS pressure is less than 1000 psig,
(2) Offsite integrated dose approaches 500 mRem CDE (Thyroid)

D. (1) OTSG Pressure is less than 1000 psig,
(2) Offsite integrated dose approaches 500 mRem CDE (Thyroid)

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-EOP-005, the isolation criteria is as follows: RCS pressure <1000 psig and OTSG Level > 85%, RCS pressure < 1000 psig and offsite dose projections approaches 500 mRem TEDE or 1500 mRem CDE Thyroid, RCS pressure < 1000 psig and BWST Level < 22 ft; (2) An OTSG is not isolated with RCS pressure above 1000 psig to preclude possibly lifting the MSSV.

| | | |
|---|---|---|
| A. | (1) RCS pressure is less than 1000 psig,<br>(2) BWST level < 22 feet | CORRECT: See above |
| B. | (1) OTSG pressure is less than 1000 psig,<br>(2) BWST level < 22 feet | INCORRECT: Plausible since OTSG pressure < 1000 psig would be below all setpoints for OSTG steam safety valves. Incorrect because OP-TM-EOP-005 directs isolation only after RCS pressure has lowered to less than 1000 psig. |
| C. | (1) RCS pressure is less than 1000 psig,<br>(2) Offsite integrated dose approaches 500 mRem CDE (Thyroid) | INCORRECT: Plausible since RCS Pressure < 1000 psig is correct and if the examinee believes that the limit for CDE is 500 mRem. Incorrect because the limit is 5000 mRem for TEDE. |
| D. | (1) OTSG Pressure is less than 1000 psig,<br>(2) Offsite integrated dose approaches 500 mRem CDE (Thyroid). | INCORRECT: Plausible if the examinee believes that the limit for CDE is 500 mRem. Incorrect because the limit is 5000 mRem for TEDE. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 038 | EA2.01 |
| | Importance Rating | 4.1 | 4.7 |

K/A: Steam Generator Tube Rupture: Ability to determine or interpret the following as they apply to a SGTR: When to isolate one or more S/Gs

Proposed Question: RO Question # 73

Technical Reference(s): OP-TM-EOP-005, Rev 9 OP-TM-EOP-0051, Rev 6

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-005-PCO-4

Question Source: Bank #

Modified Bank #

New X

| | | | |
|---|---|---|---|
| Question History: N/A | | Last NRC Exam: N/A | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | b.5 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinee must have knowledge of when to isolate an OTSG based on RCS pressure and BWST level.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**74** **ID: 1740715** **Points: 1.00**

Plant Conditions:

- Rx vessel head being de-tensioned.
- "A" Decay Heat Removal (DHR) train in service.
- "B" DHR train in standby.

Event:

- "A" Decay Closed Cooling Tank, DC-T-1A, rising at rate equivalent to 5 gpm.
- "A" Decay Closed System Inline Rad Monitor, RM-L-2, counts have risen.

Based on the conditions above which ONE of the below actions must be taken?

A. Isolate DC-T-1A vent.

B. Initiate Containment Isolation.

C. Swap cooling to "B" DHR, and isolate the "A" DHR.

D. Transfer heat removal to the OTSGs and secure DHR.

Answer: C

**Answer Explanation**

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) OP-TM-AOP-060, LEAKAGE WHILE ON DECAY HEAT REMOVAL, entry criteria is met; (2) The operating DHR train is leaking into the Decay Closed train; (3) The mitigation strategy is to pick up cooling on the standby train (DHR 'B' train), and then isolate the leaking train and place it in the standby mode.

| | | |
|---|---|---|
| A. | Isolate DC-T-1A vent. | INCORRECT: Plausible because this would isolate the leak from the DC train to the Auxiliary Building. Incorrect because this would not stop the leak from the DHR train to the DC train. |
| B. | Initiate Containment Isolation. | INCORRECT: Plausible because there is an RCS leak. Incorrect because containment isolation is not required . |
| C. | Swap cooling to "B" DHR, and isolate the "A" DHR. | Correct Answer: See above |
| D. | Transfer heat removal to the OTSGs and secure DHR. | INCORRECT ANSWER: This distractor is plausible because if the 'B' DHR train were not available, the crew may use plant procedures to transfer heat removal to the OTSGs. For the current plant condition, this is not the correct answer. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | 3 | |
| Group # | 4 | |
| K/A # | 2.4.9 | |
| Importance Rating | 3.8 | 4.2 |

K/A: Knowledge of low power/shutdown implications in accident (e.g. loss of coolant accident, or loss of residual heat removal) mitigation strategies.

Proposed Question: RO Question #74

Technical Reference(s): OP-TM-AOP-060, Rev 8

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-60-PCO-4

Question Source: Bank # 567293

Modified Bank #

New

Question History: Sim exam 9 Last NRC Exam: None

Question Cognitive Level: Memory or Fundamental Knowledge

| | | | |
|---|---|---|---|
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | b.11 | |
| | 55.43 | | |

Comments:

KA Match: This question matches the KA because the examinees must know the mitigation strategy for a loss of RCS water when on Decay Heat Removal.

High Cog: This question is high cog because the examinee must identify the entry criteria into OP-TM-AOP-060, and then understand the mitigation strategy to correctly provide decay heat removal.

OP-TM-AOP-060- Entry Criteria:
Decay Heat Removal is providing core cooling, and any of the following conditions exist:

- Continuous makeup > 1 gpm is required to maintain level
- *Leakage from DH, RCS, or fuel transfer canal > 1gpm is detected or observed*

Step 3.10- VERIFY the leak is in containment- It is not
RNO- GO TO Section 4.0, "Leak in Auxiliary Building or Fuel Handling Buildings".
Step 4.6- If DHR train A is "In Service" and either conditions exists:

- Leakage exists on DHR train A
- *DC A surge tank level is rising*

then place DHR train B in service as follows:

- *Initiate OP-TM-212-112, "Shifting DH Train B From DHR Operating Mode".*
- *Initiate OP-TM-212-151, "Shifting DH Train A from DHR to Standby Mode".*
- *When DH-P-1A is Shutdown, then close DH-V-12A*
- *Close DH-V-38A*

A. Distractor 1- Isolate "A" DCCW tank vent- Incorrect, this would not isolate the leak. plausible for Section 3.8 isolate leak however this wont isolate the DHR to DCCW leak, only prevent its overflow.

B. Distractor 2- Initiate Containment Isolation- Incorrect, containment isolation is not needed for this leak. Plausible for continuing in section 3 after verify leak is in containment step, need to do RNO.

C. Distractor 3- Transfer heat removal to the OTSGs and Secure DHR- We are below the point where OTSGs would be useful to remove heat. Plausible for step 3.13, however RCS is not available and this section is skipped by leak NOT in containment.

D. Correct Answer- Swap cooling to "B" DHR and close DH-V-12A and DH-V-38A- See above. Swap cooling is correct per section 4.5 of AOP-060.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**75** **ID: 1736802** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Which on of the following identifies the status of the Reactor Building Hydrogen Monitor Recorders?

They are both __(1)__ and they are located on Control Room __(2)__.

A. (1) ON (2) the H&V Panel.

B. (1) ON (2) Panel Left.

C. (1) OFF (2) the H&V Panel.

D. (1) OFF (2) Panel Left.

Answer: D

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) In accordance with 1105-18, CONTAINMENT HYDROGEN MONITOR, the H2 Monitors are normally maintained in the standby condition (isolated from containment but with the electronics kept warm), the control room chart recorder drive will remain off and will be turned on when the monitor is placed in service. (2) During normal plant operations, the Hydrogen Monitor System will be warmed up and maintained in a Standby mode. (3) Containment Hydrogen Monitoring are placed into service when required per procedure. (4) Recorder are Located on Control room Panel Left

| | | |
|---|---|---|
| A. | (1) ON (2) H&V Panel | INCORRECT: Plausible because the the Hydrogen Monitors are maintained in Standby and the examinee may think the Recorder are ON. H&V Panel has many Ventilation Controls and other monitors but not the H2 Monitor Controls. |
| B. | (1) ON (2) Panel Left. | INCORRECT: Plausible because the the Hydrogen Monitors are maintained in Standby and the examinee may think the Recorder are ON. Correct Location |
| C. | (1) OFF (2) H&V Panel | INCORRECT: Plausible since they are OFF but wrong location. |
| D. | (1) OFF (2) Panel Left | CORRECT: See above |

| Examination Outline Cross-reference: | Level | RO | | SRO |
|---|---|---|---|---|
| | Tier # | 2 | | |
| | Group # | 2 | | |
| | K/A # | 028 | A4.03 | |
| | Importance Rating | 3.1 | | 3.3 |

K/A: Hydrogen Recombiner and Purge Control System: Ability to manually operate and/or monitor in the control room: Location and operation of hydrogen sampling and analysis of containment atmosphere, including alarms and indications.

Proposed Question: RO Question # 75

Technical Reference(s): 1105-18, Rev 14

Proposed References to be provided to applicants during examination: None

Learning Objective: 240-GLO-10

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

| | | |
|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X |
| | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.41 | b.7 |
| | 55.43 | |

Comments:

KA Match: This question matches the KA because the examinee must know the location of and operation of the Hydrogen Monitors and their indications. The Hydrogen Monitors are monitored in the Control Room by recorders, the examinee should know the status of the H2 Monitors including that their recorders are NOT on during normal operation. Operator action in the Control Room and in the plant is required to place the H2 Monitors in operation.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**76** **ID: 1737249** **Points: 1.00**

Plant Conditions:

- Reactor has tripped on RCS low pressure.
- Pressurizer level is 370 inches and slowly increasing.
- RCS pressure is 1500 psig and slowly lowering.
- Core exit temperature is 556 degrees F and stable.
- Reactor Building sump level is 38 inches and increasing.

(1) What is the location of the RCS Leak?
(2) Which of the following combinations of procedures provide the correct guidance to cooldown the plant?

A. (1) Leak in the RCS Cold Leg
(2) Enter OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN, and use OP-TM-EOP-006, LOCA COOLDOWN to cooldown the plant.

B. (1) Leak in the RCS Cold Leg
(2) Enter OP-TM-AOP-050, REACTOR COOLANT LEAKAGE and use 1102-11, PLANT COOLDOWN to cooldown the plant.

C. (1) Leak in the Pressurizer Steam Space
(2) Enter OP-TM-AOP-043, LOSS OF PRESSURIZER, and use 1102-11, PLANT COOLDOWN to cooldown the plant.

D. (1) Leak in the Pressurizer Steam Space
(2) Enter OP-TM-AOP-050, REACTOR COOLANT LEAKAGE and use 1102-11, PLANT COOLDOWN to cooldown the plant.

Answer: C

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) To answer this question correctly, the examinee must determine that a leak in the Pressurizer Steam Space exists; (2) In the stem, it is clear that RCS leakage exists, but to differentiate this Pressurizer Steam Space leak from an leak in the RCS cold leg the examinee will have to deduce that RCS pressure is LOW for the given Pressurizer level and RCS Temperature; (3) If this were a RCS cold leg leak, for a pressure this low, pressurizer level would be much lower as well; (4) For the parameters given, the correct procedure the crew must implement is EOP-001, then enter AOP-043 and 1102-11 to cooldown the plant.

| | | |
|---|---|---|
| A. | (1) Leak in the RCS cold leg (2) Enter OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN, and use OP-TM-EOP-006, LOCA COOLDOWN to cooldown the plant. | INCORRECT: Plausible because the examinee could believe there is a leak other than an pressurizer steam space leak by RCS pressure lowering and RB Sump level rising. In addition, the examinee will have to check if conditions exist for the operating crew to enter OP-TM-EOP-002 for a Loss of SCM (They do not). Within that procedure, if HPI were not adequate to maintain pressurizer level the crew would enter OP-TM-EOP-006 to cooldown the plant. |
| B. | (1) Leak in the RCS cold leg (2) Enter OP-TM-AOP-050, REACTOR COOLANT LEAKAGE and use 1102-11, PLANT COOLDOWN to cooldown the plant. | INCORRECT: Plausible because the examinee could believe there is a leak other than an pressurizer steam space leak by RCS pressure lowering and RB Sump level rising. In addition, if the examinee does not recognize that the symptoms for a Loss of SCM exist, then AOP-050 could be entered. In AOP-050, 1102-11 is used to cooldown the plant. |
| C. | (1) Leak in the Pressurizer steam space. (2) Enter OP-TM-AOP-043, LOSS OF PRESSURIZER, and use 1102-11, PLANT COOLDOWN to cooldown the plant. | CORRECT ANSWER: OP-TM-AOP-043 must be entered, and then perform 1102-11 with guidance from an attachment in AOP-043. |
| D. | (1) Leak in the Pressurizer steam space. (2) Enter OP-TM-AOP-050, REACTOR COOLANT LEAKAGE and use 1102-11, PLANT COOLDOWN to cooldown the plant. | INCORRECT: Plausible because a leak in the Pressurizer Steam Space is correct. Incorrect because OP-TM-AOP-050 does not provide the additional guidance required to cooldown the plant with a leak in the pressurizer steam space. OP-TM-AOP-043 provides additional guidance on the low SCM, and solid plant operations. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 008 | AA2.12 |
| | Importance Rating | 3.4 | 3.7 |

K/A: Pressurizer Vapor Space Accident: Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: PZR level indicators

Proposed Question: Question #76

Technical Reference(s): OP-TM-AOP-043, Rev 6

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

OP-TM-AOP-0431, Rev 5

Proposed References to be provided to applicants during examination: Steam Tables

Learning Objective: AOP-043-PCO-4

Question Source: Bank #

Modified Bank # 462810

New

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 b.5

Comments:

KA Match: This question matches the KA because the examinee must use the Pressurizer Level Indicator to determine that there is a leak in the Pressurizer Steam Space, vice a leak elsewhere in the RCS. The pressurizer level is rising, while pressure is lowering which is indicative of a pressurizer steam space leak.

High Cog: This question is high cog because the examinee must analyze the conditions in the stem to determine the location of the leak.

SRO ONLY: This question is SRO only because the examinee must assess the plant conditions, and then select and determine the required course of action.

Plant Conditions:

- Reactor has tripped on RCS low pressure.
- Pressurizer level is 240 inches and increasing.
- RCS pressure is constant at 1100 psig.
- Core exit temperature is 556 degrees F and stable.
- Reactor Building sump level is 38 inches and increasing.

From the list below, identify the ONE transient in progress.

A. Leak in the RCS cold leg.

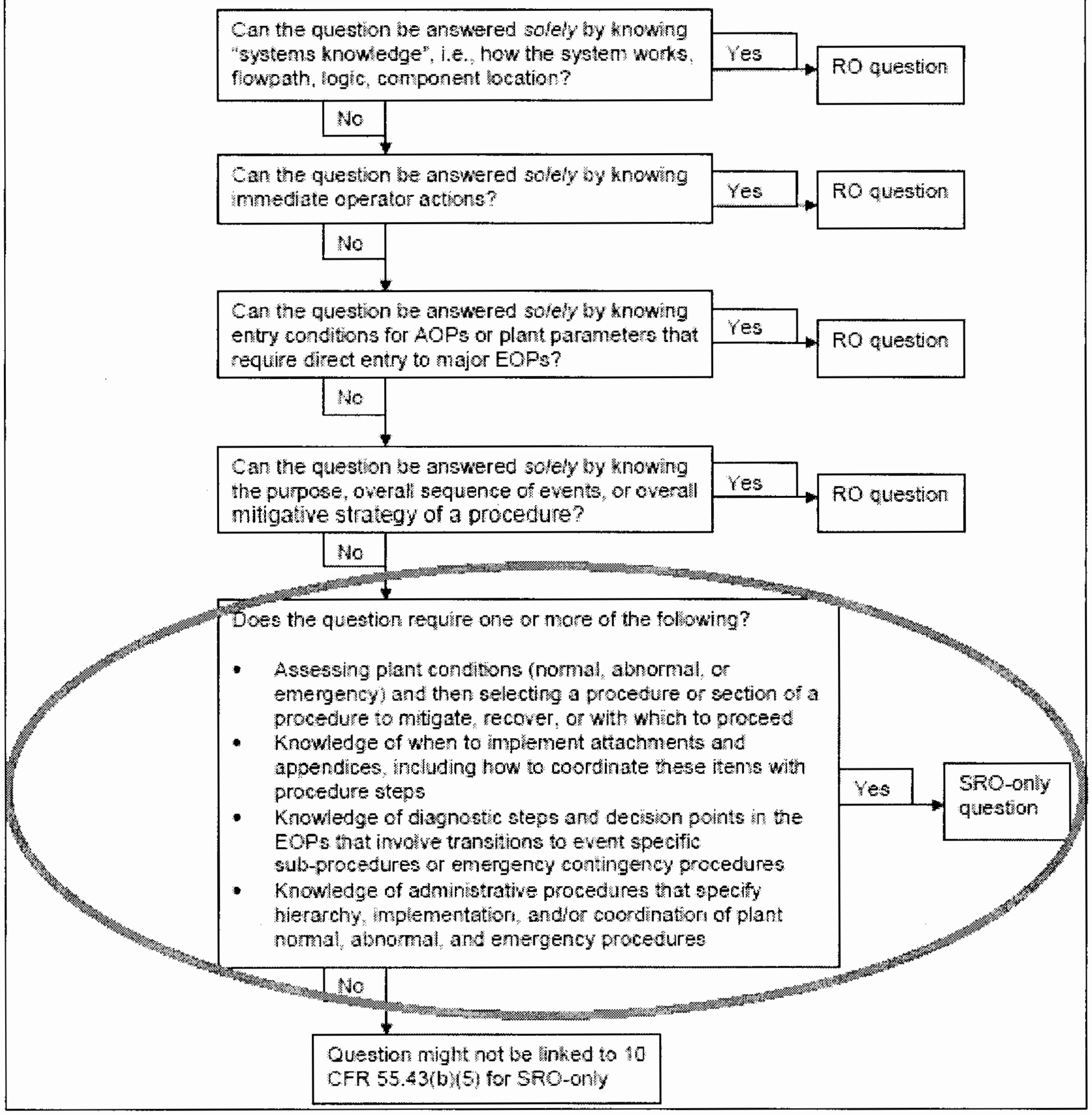
B. Leak in the Pressurizer steam space.

C. Rx vessel head vent valves failed open.

D. Failure of the Pressurizer level control valve (MU-V-17) to close.

Answer: B

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

# EXAMINATION ANSWER KEY



**77** **ID: 1700250** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- MAP C-3-2, IC SURGE TANK LEVEL HI/LO, alarm in.
- RM-L-9, Intermediate Closed Cooling Water Radiation Monitor, counts are rising.
- IC-T-1, Intermediate Closed Cooling Water Surge Tank, level is off-scale high, and an attempt to bring IC-T-1 back on scale has failed.
- OP-TM-AOP-050, REACTOR COOLANT LEAKAGE is entered.

Current plant parameters:

- Pressurizer level is 220 inches and steady.
- MU Tank level is 86 inches and slowly lowering.
- Reactor Building pressure is 0.2 psig and steady.

Which ONE of the following choices identifies the correct procedure and actions that must be implemented?

A. 1) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
2) Initiate a plant shutdown to be in HOT SHUTDOWN within 24 hours

B. 1) OP-TM-MAP-C0302, IC SURGE TANK LEVEL HI/LO
2) Trip the reactor and close IC-V-2 and IC-V-3

C. 1) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE
2) Trip the reactor and close IC-V-2 and IC-V-3

D. 1) OP-TM-MAP-C0302, IC SURGE TANK LEVEL HI/LO
2) Initiate a plant shutdown to be in HOT SHUTDOWN within 24 hours

Answer: B

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<<<Explanation: To answer this question correctly, the examinee must know: (1) With the indication of the IC-T-1 level high, RM-L-9 rising, the MU Tank level lowering with pressurizer level remaining steady, there is a small break leak between the RCS and ICCW system (either the RCP seals, or Letdown coolers); (2) Because RM-L-9 is rising and IC-T-1 is already off-scale high, the actions to mitigate this event are to trip the reactor, trip all four reactor coolant pumps, close IC-V-2 and 3 and initiate OP-TM-AOP-050 "RCS Leakage", which is found in OP-TM-MAP-C0302.

| | | |
|---|---|---|
| A. | 1) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE<br>2) Initiate a plant shutdown to be in HOT SHUTDOWN within 24 hours | Plausible because the examinee should determine that and RCS leak is in progress. Actions to mitigate RCS leaks (including into ICCW) are found in OP-TM-AOP-050, but this answer is incorrect because this procedure routes the examinee to OP-TM-MAP-C0302 IC Surge Tank Level Hi/Lo. In addition, in normal procedure hierarchy the mitigation steps of this nature would be found in an AOP or an EOP. AOP-050 directs the a plant shutdown to HOT SHUTDOWN. This is also incorrect because OP-TM-MAP-C0302 directs a reactor trip. |
| B. | 1) OP-TM-MAP-C0302, IC SURGE TANK LEVEL HI/LO<br>2) Trip the reactor and close IC-V-2 and IC-V-3 | Correct Answer - The actions to mitigate an RCS to ICCW leak, where IC-T-1 cannot be maintained on scale is step 4.1.3 of this alarm response procedure. The examinee could arrive at this answer by directly entering the alarm response or OP-TM-AOP-050, step 3.5 which directs the examinee to OP-TM-MAP-C0302. Most primary mitigation steps are not found in alarm response procedures. |
| C. | 1) OP-TM-AOP-050, REACTOR COOLANT LEAKAGE<br>2) Trip the reactor and close IC-V-2 and IC-V-3 | Plausible because the examinee should determine that and RCS leak is in progress. Actions to mitigate RCS leaks (including into ICCW) are found in OP-TM-AOP-050, but this answer is incorrect because this procedure routes the examinee to OP-TM-MAP-C0302 IC Surge Tank Level Hi/Lo. In addition, in normal procedure hierarchy the mitigation steps of this nature would be found in an AOP or an EOP. AOP-050 directs the a plant shutdown to HOT SHUTDOWN. This is also incorrect because OP-TM-MAP-C0302 directs a reactor trip. |
| D. | 1) OP-TM-MAP-C0302, IC SURGE TANK LEVEL HI/LO<br>2) Initiate a plant shutdown to be in HOT SHUTDOWN within 24 hours | Plausible because part one is correct. Incorrect because the alarm response directs the examinee to trip the reactor and close IC-V-2 and 3. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 009 | 2.4.31 |
| | Importance Rating | | 4.1 |

K/A: Small Break LOCA: Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: Question #77

Technical Reference(s): OP-TM-MAP-C0302, Rev 3; OP-TM-EOP-010, Rev 19; OP-TM-AOP-050, Rev 6; OP-TM-AOP-032, Rev 4

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Proposed References to be provided to applicants during examination: | | | None |
| Learning Objective: | AOP-050-PCO-1 | | |
| Question Source: | Bank # | | |
| | Modified Bank # | | |
| | New | X | |
| Question History: | N/A | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | b.5 | |
| Comments: | | | |

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

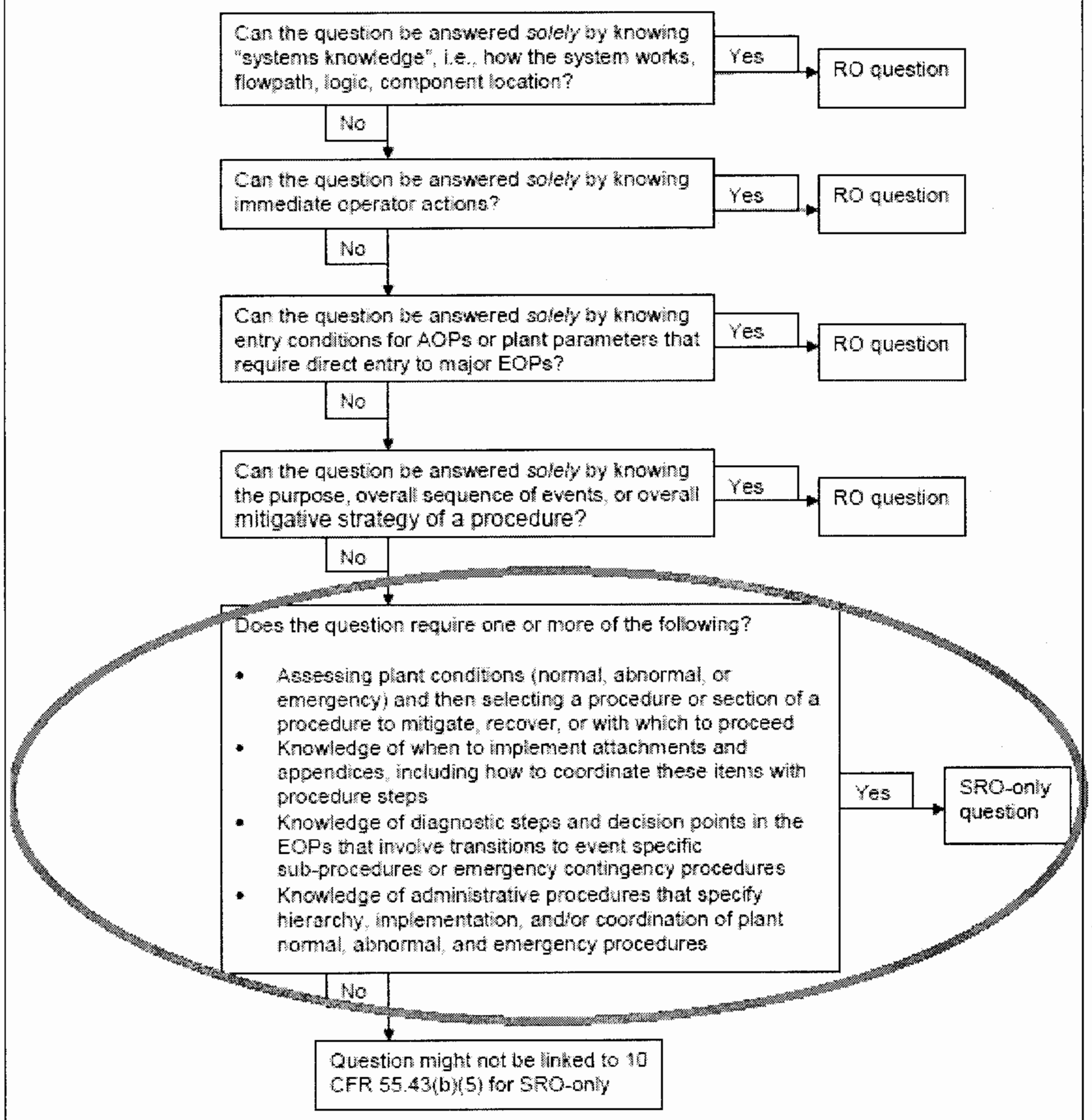
K/A MATCH: This matches the K/A because the examinee must know that the alarm response in this question provides the mitigating steps for the RCS leak into the ICCW system.

High Cog: This examinee must determine that a small RCS leak exists, and that the RCS is leaking into the ICCW stem.

SRO Only: The question requires the examinee to assess plant conditions and to know the content of the listed procedures in order to select a required course of action. These procedures are not major EOPs.

**ES-401** **8** **Attachment 2**

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**78** **ID: 1685279** **Points: 1.00**

Plant conditions:

- Reactor is at 100% power
- DC-P-1A is tagged out for a bearing replacement.
- MU-P-1A and MU-P-1C are ES selected and operable.
- The Secondary NLO has just reported the fuel rack for the SBO Diesel was accidentally broken in the **tripped position** by a contract worker climbing over the lever.
- Maintenance says the repair will take 4 hours.

EVENT:

- One-half of the overhead lights go out, and do NOT come back on.
- You observe the following MAPs actuate:
  - B-1-1, 4KV ES FDR BKR TRIP.
  - B-2-1, 4KV ES BUS UV/OV.
  - B-1-2, 4KV ES MOTOR TRIP.
  - F-1-5, RCP SEAL TOT INJECT FLOW HI/LO.

Based on these conditions, which ONE of the following actions must be taken?

A. Initiate Reactor shutdown within 1 hour.

B. Swap MU-P-1A Cooling from DC to NS IAW OP-TM-543-439.

C. Initiate OP-TM-864-901 SBO Diesel Generator (EG-Y-4) Operations.

D. Ensure Emergency Diesel Generator EG-Y-1B restored to operable status within 7 days.

Answer: A

**Answer Explanation**

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) When one-half of the overhead lights go off, and with the MAP B alarms present with MAP F-1-5, that the 1E 4160V bus has been lost; (2) With a loss of the 1E 4160V bus, and DC-P-1A out of service that the plant is in TS 3.3.1.1.d; (3) The action statement when TS 3.3.1.1.d is met is to apply TS 3.0.1 to initiate a reactor shutdown within 1 hour.

| | | |
|---|---|---|
| A. | Initiate Reactor shutdown within 1 hour. | Correct Answer |
| B. | Swap MU-P-1A Cooling from DC to NS IAW OP-TM-543-439 | Plausible because this is an action the crew would normally take if Seal Injection were lost and MU-P-1A needed to be started. In this case, Seal Injection is lost, but NSCCW is already lined up to MU-P-1A. This would have happened when DC-P-1A was tagged out for maintenance. |
| C. | Initiate OP-TM-864-901 SBO Diesel Generator (EG-Y-4) Operations. | Plausible because use of the SBO Diesel is desired, but the SBO is not available per the question stem. |
| D. | Ensure Emergency Diesel Generator EG-Y-1B restored to operable status within 7 days. | Plausible because this Tech Spec (TS 3.7) would apply since EG-Y-1B did not start. Incorrect because TS 3.3.1.1d is met and the crew has to initiate a shutdown within 1 hour. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 026 | AA2.02 |
| | Importance Rating | | 3.6 |

K/A: Loss of Component Cooling Water: Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: The cause of possible CCW loss

Proposed Question: Question #78

Technical Reference(s): T.S 3.3, AMD 289

T.S 3.7,

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-014-PCO-1

Question Source: Bank # 503953

Modified Bank #

New

# EXAMINATION ANSWER KEY



Question History: N/A Last NRC Exam: N/A

| | | |
|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | |
| | Comprehension or Analysis | X |
| 10 CFR Part 55 Content: | 55.41 | |
| | 55.43 | 5 |

Comments:

K/A Match: This matches the K/A because the examinee has to determine that both of the decay closed cooling water pumps are inoperable, hence Decay Closed Cooling is lost.

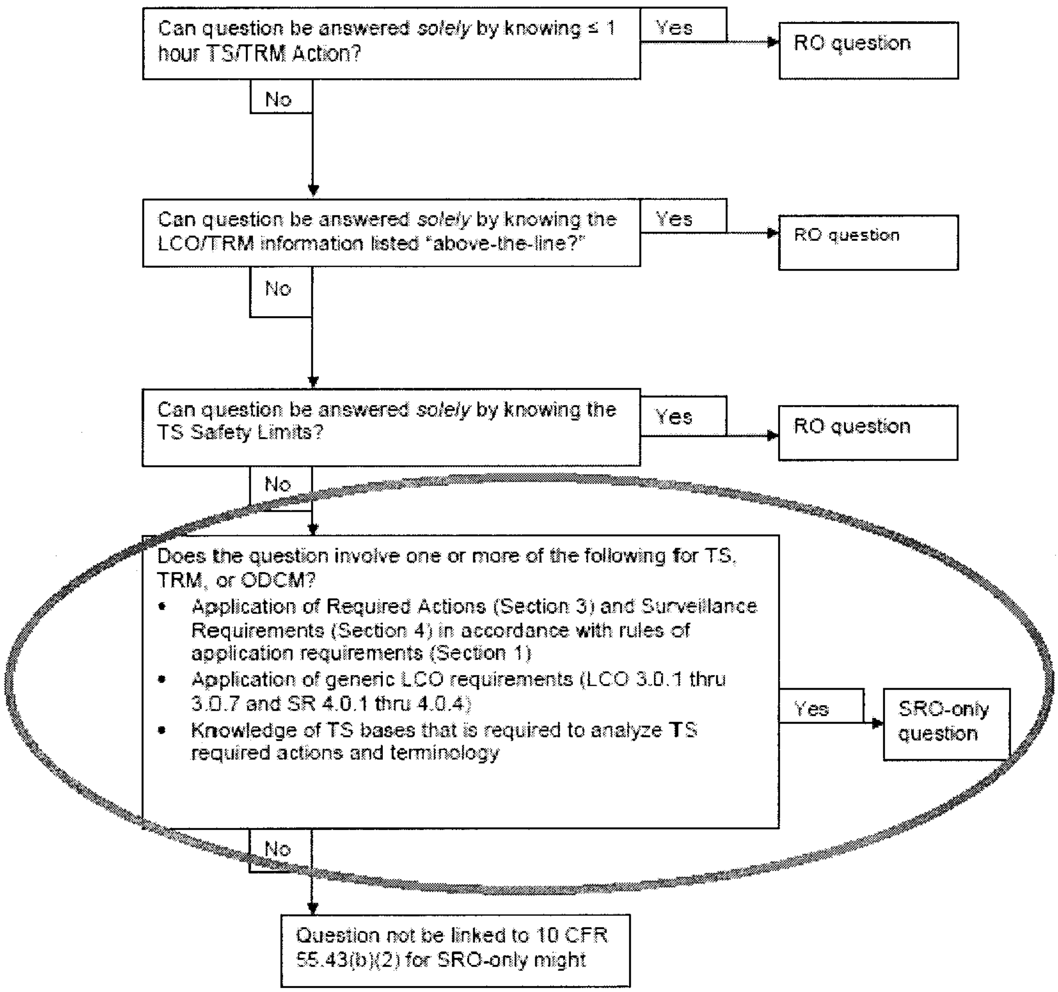
High Cog: This is high cog because the examinee must analyze the power loss and come to the understanding that both decay closed cooling water pumps are lost, then realize the plant does not me the requirements of TS 3.4.1.1.b. The examinee must also know the action for no decay closed cooling water pumps is to initiate a reactor shutdown within 1 hour.

SRO Only: This is an SRO only question because if involves the application of T.S. 3.0.1. The SRO must analyze the stem, and realize that DC-P-1B is not operable or performing it's ES function (due to EG-Y-1B not powering the 1E 4160V bus) and apply T.S. 3.0.1.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)**

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# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**79** **ID: 1685296** **Points: 1.00**

Plant conditions:

- Reactor power is 35%
- Turbine load is 300 MWe.
- Main Feedwater Pump 1A (FW-P-1A) is tagged out of service.
- All other equipment lineups are normal.

Event:

- MAP Alarm M-1-7, FWP-1B TRIP, actuates.
- Reactor power and turbine load are stable.
- OTSG “A” and “B” levels are lowering.

Given the above information, which ONE of the following actions is required?

A. Initiate a MANUAL reactor trip and enter OP-TM-EOP-001, REACTOR TRIP.

B. Ensure MAP H-1-1, ICS RUNBACK, actuates and implement 1102-4, POWER OPERATIONS.

C. Rapidly reduce main turbine load until reactor power is less than 7% and implement OP-TM-424-901, EMERGENCY FEEDWATER.

D. Trip the main turbine and reduce reactor power to less than 7% and then stabilize the plant in accordance 1102-4, POWER OPERATIONS.

Answer: A

**Answer Explanation**

<<<<Explanation: To answer this question correctly, the examinee must know: (1) When reactor power is above 7%, the Main Feedwater Pump trip (where at least on MFW pump must be running) is active, meaning if reactor power is above 7% and no MFW pumps are running, the reactor and turbine should trip; (2) The presence of M-1-7, with OTSG levels lowering, but reactor power and turbine loads stable is indicative that the reactor did not trip when FW-P-1B tripped (FW-P-1A was tagged out of service in the stem, and reactor power is 35%); (3) The action for that is to trip the reactor and enter OP-TM-EOP-001, REACTOR TRIP.

| | | |
|---|---|---|
| A. | Initiate a MANUAL reactor trip and enter OP-TM-EOP-001, REACTOR TRIP. | Correct Answer |
| B. | Ensure MAP H-1-1, ICS RUNBACK, actuates and implement 1102-4, POWER OPERATIONS. | Plausible because at a higher power, when a MFW pump is lost an runback would occur (if both MFW pumps were running). In addition when both MFW pumps are lost, EFW actuates, which would provide a source of feedwater, although not one that is acceptable at 35% power to runback to less than 7% power. |
| C. | Rapidly reduce main turbine load until reactor power is less than 7% and implement OP-TM-424-901, Emergency Feedwater. | Plausible because at the point where the question ends, there is no other indication that any RCS parameter (i.e. RCS pressure, or reactor power) is exceeding an RPS trip setpoint. The examinee could determine since the reactor power and turbine load are stable that performing a plant runback is an acceptable option. |
| D. | Trip the main turbine and reduce reactor power to less than 7% and then stabilize the plant in accordance 1102-4, POWER OPERATIONS. | Plausible because this is the below the RPS automatic trip interlock for Loss of Both Main Feedpumps. Incorrect because the examinee must know that the reactor must be tripped when both main feedpumps are lost at 35% power. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 054 | AA2.01 |
| | Importance Rating | | 4.4 |

K/A: Loss of Main Feedwater: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Occurance of reactor and/or turbine trip.

Proposed Question: Question #79

Technical Reference(s): OP-TM-MAP-M0107, Rev 1

OP-TM-EOP-001

Proposed References to be provided to applicants during examination: None

Learning Objective: 401-GLO-11

Question Source: Bank # 1044555

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | |
|---|---|---|
| | Modified Bank # | |
| | New | |
| Question History: | Systems Exam 14 Last NRC Exam: | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | |
| | Comprehension or Analysis | X |
| 10 CFR Part 55 Content: | 55.41 | |
| | 55.43 | 5 |

Comments: (KA Match and why high cog)

K/A Match: This is a match because the examinee has to interpret that at reactor trip should have occurred when the only operating MFW pump tripped.

High Cog: This question is high cog because the examinee has to analyze the event to understand that a MFW pump did trip, but the reactor did not trip. The examinee has to determine the correct course of action for the operating crew to take.

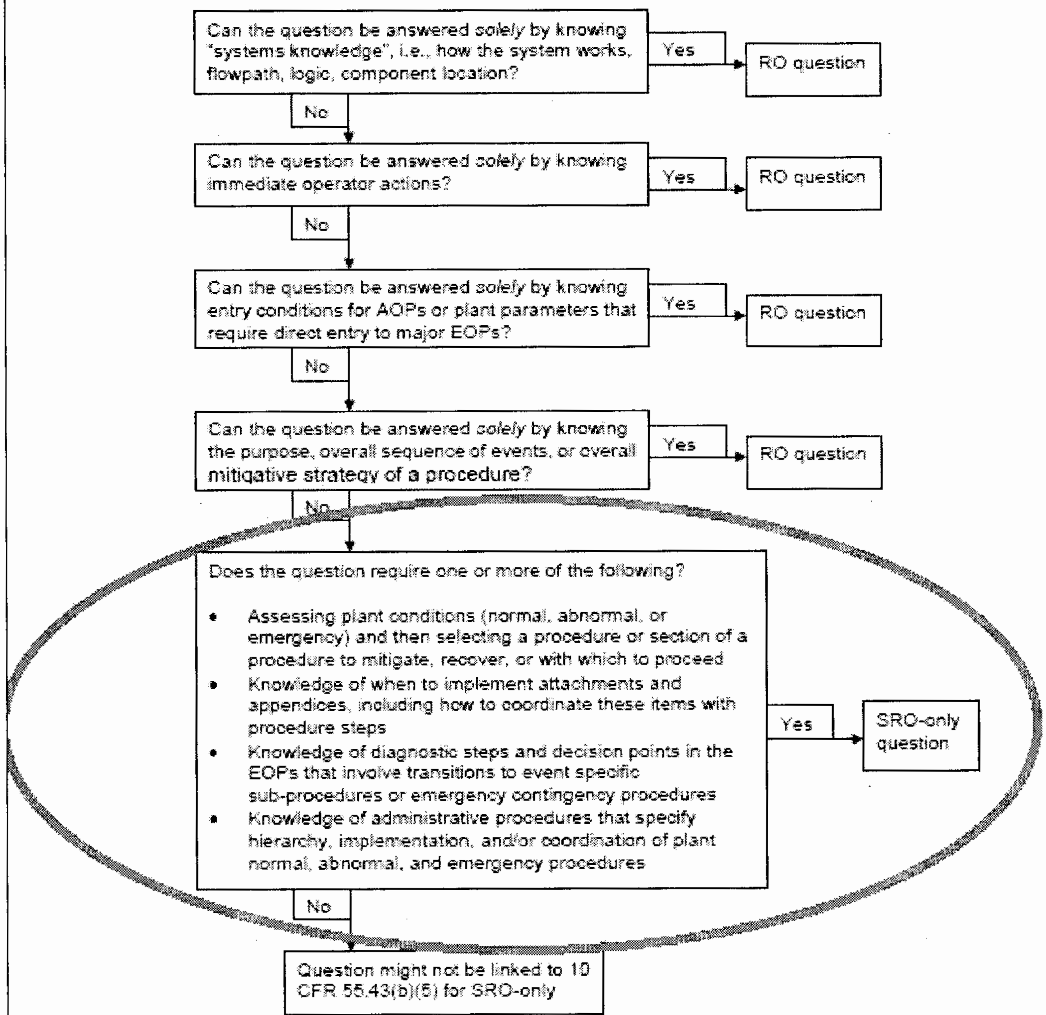
SRO ONLY: This question is SRO only because it requires the SRO examinee to determine that an ATWS has occurred. The SRO examinee must choose between the various procedures which will shutdown the plant.

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM



**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)**
**(Assessment and selection of procedures)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**80** **ID: 1685310** **Points: 1.00**

Plant Conditions:

- RCS Heat up in progress.
- RCS Temperature is 290°F and rising.
- DC Distribution Panel 1M transfer switch is selected to "A" DC Distribution System and the switch on PCR is in "Auto".
- Makeup Pump 1B is supplying seal injection.

Event:

- The following alarms have actuated simultaneously:
  - AA-3-2 7 KV Bus Trouble.
  - AA-3-3 4 KV BOP Bus Trouble.
  - AA-3-5 480V BOP Bus Trouble.
  - A-1-7 Battery 1A Discharging.
  - A-2-7 Battery Charger 1A/1C/1E Trouble.
  - A-3-7 Inverter 1A/1C/1E Trouble.
  - B-3-1 4KV ES Bus Trouble.
  - PRF1-1-1 CRDM Brkr Test Trouble.
  - H&V A 4-2 Contr Bldg Bat Chgrs A Damper Tbl Fire-Smoke.
  - Loss of numerous breaker status lights at control switches.

Based on these conditions, identify the ONE selection below that describes the:
(1) Applicable procedure to respond to these conditions, and
(2) Impact on the ability to operate safety related equipment.

A. (1) OP-TM-AOP-023, "A" DC System Failure.
(2) Makeup Pump 1B breaker must be racked out due to LTOP concerns.

B. (1) OP-TM-AOP-023, "A" DC System Failure.
(2) Pressurizer Level must be verified ≤ 100 inches because the PORV setpoint is greater than 592 psig.

C. (1) OP-TM-AOP-024, "B" DC System Failure.
(2) Makeup Pump 1B breaker must be racked out due to LTOP concerns.

D. (1) OP-TM-AOP-024, "B" DC System Failure.
(2) Pressurizer Level must be verified ≤ 100 inches because the PORV setpoint is greater than 592 psig.

Answer: B

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) Indications from the event are representative to a Loss of 'A' DC; (2) that the PORV, RC-RV-2 is not operable when there is no 'A' DC; (3) Because there the plant is below 313F, then the requirements of Tech Spec 3.1.12 must met; (4) At the point before the loss of 'A' DC, and in accordance with 3.1.12.1, since there is an HPI breaker racked in (MU-P-1B), then MU-V-16A-D must be closed with breakers open, and MU-V-217 must be closed. In addition pressurizer level must be less than 100 inches; (5) When the PORV becomes inoperable after the loss of 'A' DC then the operating crew has 2 options to do within 8 hours: restore the setpoint below the maximum value (i.e. restore 'A' DC); or verify pressurizer level is ≤ 100 inches and satisfy the requirements of T.S 3.1.12.3. (6) Having MU-V-16A-D closed with their breakers open and MU-V-217 satisfies this requirement, and is already complete because MU-P-1B breaker is racked in supplying seal injection, hence no further action is required other than ensure pressurizer level is ≤ 100 inches.

| | | |
|---|---|---|
| A. | ((1) OP-TM-AOP-023, "A" DC System Failure.<br>(2) Makeup Pump 1B breaker must be racked out due to LTOP concerns. | Plausible because 'A' DC was lost, and racking out the makeup pump would be an option. Incorrect because the makeup pump does NOT have to be racked out. Since MU-V-16A-D are already closed with breakers racked out, and MU-V-217 is closed because the HPI breaker is racked in, the only action the crew would have to take is to ensure pressurizer level is ≤ 100 inches. |
| B. | (1) OP-TM-AOP-023, "A" DC System Failure.<br>(2) Pressuizer Level must be verified ≤ 100 inches because the PORV setpoint is greater than 592 psig. | Correct Answer. |
| C. | (1) OP-TM-AOP-024, "B" DC System Failure.<br>(2) Makeup Pump 1B breaker must be racked out due to LTOP concerns. | Plausible because the examinee could believe that 'B' DC was lost. In addition the reactor coolant system, reactor vessel, and pressurizer vents fail closed. The examinee could believe that these are LTOP valves because they could be opened to lower pressure. In addition, they could believe that racking the HPI breaker out is a compensatory action for T.S. 3.1.13 requiring a vent path from a Reactor Coolant System High Point vent (RC-40B and 41B are failed closed). This is not entered because RC-V-40A and 41A are still operable. |
| D. | (1) OP-TM-AOP-024, "B" DC System Failure.<br>(2) Pressuizer Level must be verified ≤ 100 inches because the PORV setpoint is greater than 592 psig. | Plausible because the examinee could believe that 'B' DC was lost. Also, due to loss of some RCS, RV, and Pressurizer Vents, the examinee could believe that pressurizer level must be maintain less than 100 inches. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 058 | AA2.03 |
| | Importance Rating | | 5 |

K/A: Loss of DC Power: Ability to determine and interpret the following as they apply to the Loss of DC Power: DC loads lost; impact on ability to operate and monitor plant systems.

Proposed Question: Question #80

Technical Reference(s): OP-TM-AOP-023, Rev 7

T.S. 3.1.12.3, AMD 281

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-023-PCO-1

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 b 5

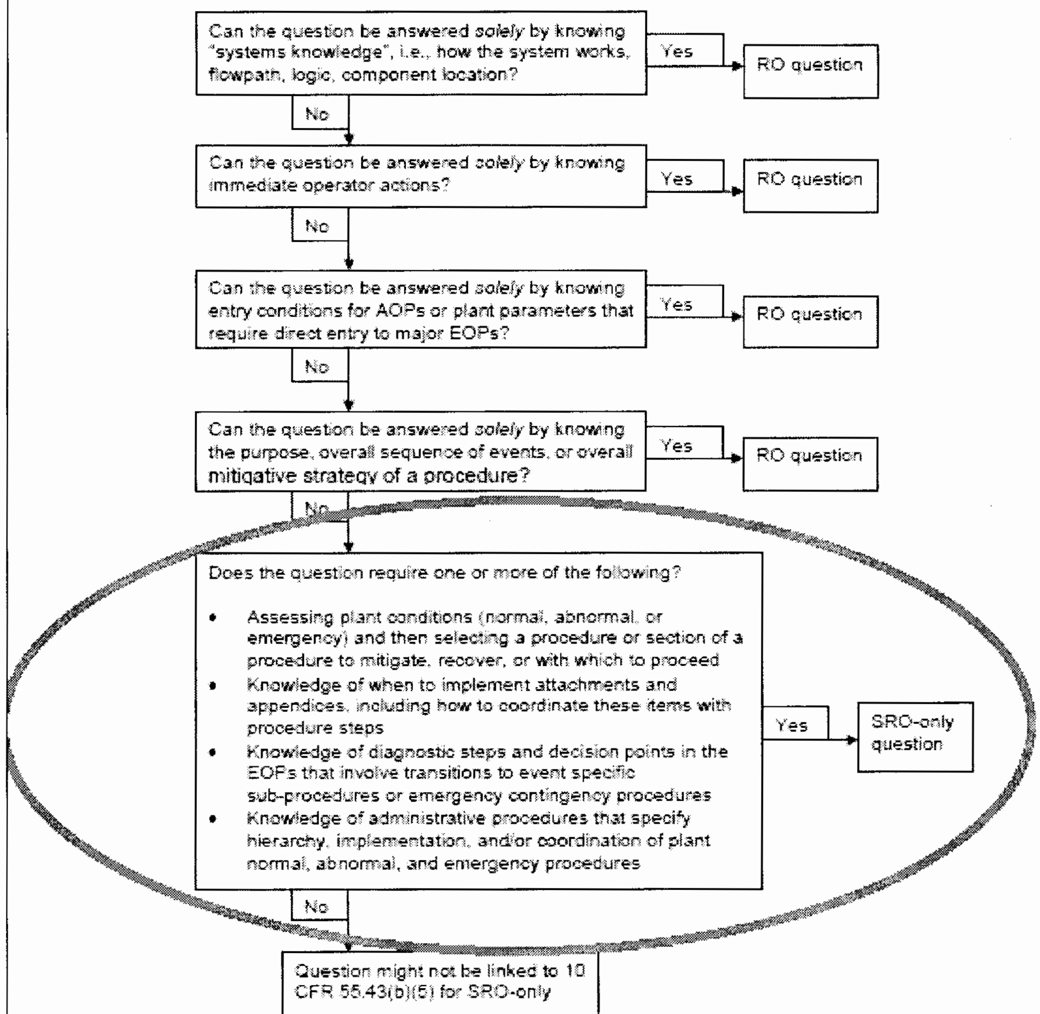
Comments:

K/A Match: This question matches the K/A because the examinee must recognize that 'A' DC is lost, then choose the correct safety related equipment that is effected by the loss.

High Cog: This is high cog question because the student must analyze the alarms in the stem, then choose correct procedure and effected safety related equipment.

SRO Only: This question is SRO only because in the second part of this question the SRO examinee must determine the PORV operability based on the loss of DC power. The PORV is inoperable on a total loss of 'A' DC, but not necessarily inoperable on a partial loss of 'A' DC. The SRO examinee will have to determine applicable portion of OP-TM-AOP-023, and then determine since it is a complete loss of 'A' DC that the PORV is inoperable based on the fact that RCS temperature is <313°F and the maximum lift setpoint is greater than 592 psig.

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)**
**(Assessment and selection of procedures)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**81** **ID: 1720494** **Points: 1.00**

Plant conditions:

- A loss of vacuum of causes both Main Feedwater pumps to trip.
- Due to equipment failures EF-P-2A, Motor Driven Emergency Feedwater Pump, is the only operating pump to supply FW to the OTSGs.
- RCS is at HSD conditions:
  - RCS T-Hot is 532 degrees F and rising.
  - RCS pressure is 2150 psig and rising.
  - OTSG pressures are 885 psig and steady.
  - OTSG levels are 25" and steady.
- FW-P-1A vacuum is being restored.

EVENT:

- EF-P-2A trip.
- Incore temperatures are rising.

Based on these conditions identify the ONE selection below that describes:
(1) Method of core cooling to be established.
(2) Applicable procedure.

A. (1) Rapid cooldown to LPI injection.
(2) OP-TM-EOP-002, LOSS OF 25 DEGREES F SUBCOOLING MARGIN.

B. (1) Condensate booster pump feed.
(2) OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER.

C. (1) Rapid cooldown to LPI injection.
(2) OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER.

D. (1) Condensate booster pump feed.
(2) OP-TM-EOP-002, LOSS OF 25 DEGREES F SUBCOOLING MARGIN.

Answer: B

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) One of the entry criteria for OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER, is met (incore temperatures rising and NO FEEDWATER available); (2) Once the EOP is entered, the examinee must know the criteria for each know the routing step that applies to go begin cooling the Reactor, for this question, the examinee must realize that a condensate pump is on, a reactor coolant pump is on, therefore Attachment 1 of OP-TM-EOP-004 is the method of cooling which should be used next.

| | | |
|---|---|---|
| A. | (1) Rapid cooldown to LPI injection.<br>(2) OP-TM-EOP-002, LOSS OF 25 DEGREES F SUBCOOLING MARGIN. | Incorrect - Plausible because OP-TM-EOP-004 mitigation strategy involves lowering OTSG pressure to strengthen the heat sink to establish heat transfer out of the OTSG's. In addition, if the examinee believes that if SCM is less than 25F, that OP-TM-EOP-002 is entered then it may be possible to perform a cooldown to LPI. |
| B. | (1) Condensate booster pump feed.<br>(2) OP-TM-EOP-004, Lack of Primary to Secondary Heat Transfer. | Correct Answer - The stem only indicates that both Main Feedwater Pumps are tripped. All equipment is available to support Condensate Booster Pump feed. |
| C. | (1) Rapid cooldown to LPI injection.<br>(2) OP-TM-EOP-004, LACK OF PRIMARY TO SECONDARY HEAT TRANSFER. | Incorrect- Plausible because in OP-TM-EOP-004, a cooldown is performed after Feedwater becomes available. Incorrect because the cooldown is not a rapid cooldown to LPI. |
| D. | (1) Condensate booster pump feed.<br>(2) OP-TM-EOP-002, LOSS OF 25 DEGREES F SUBCOOLING MARGIN. | Incorrect- Plausible because the operators must feed in accordance with Rule 4, which includes raising OTSG levels to 75% to 85% in the operating range. Incorrect because condensate booster pump cooling is not an option in that guide. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | | 1 |
| Group # | | 1 |
| K/A # | E04 | 2.2.44 |
| Importance Rating | | 4.4 |

K/A: Inadequate Heat Transfer: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question: Question #81

Technical Reference(s): OP-TM-EOP-004, Rev 11

OS-24, Rev 28

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP004-PCO-4

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question Source: | Bank # | 363689 | |
| | Modified Bank # | | |
| | New | | |
| Question History: | Comp 2 | Last NRC Exam: | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | b.5 | |

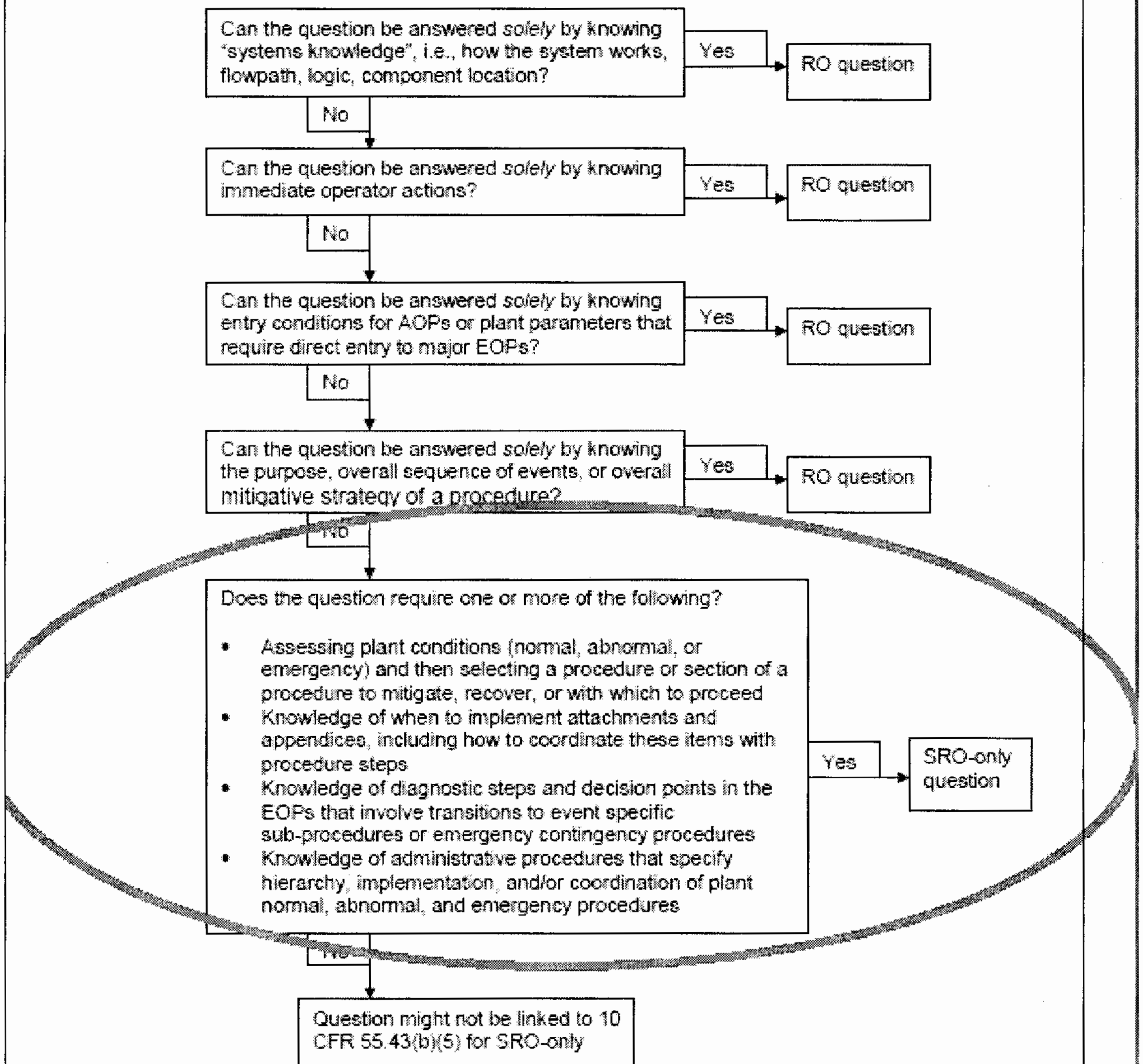
Comments:

KA Match: This question matches the KA because the examinee must interpret stem indications to determine the correct path for establishing feedwater to the OTSGs. The operator actions will establish core cooling.

High Cog: This question is High Cog because the examinee will have analyze the stem conditions to determine there is no feedwater available to the OTSGs.

SRO ONLY: This question is SRO under b.5 because in addition to recognizing the entry conditions into OP-TM-EOP-004 the examinee must assess plant conditions then select the NEXT procedure which will provide feedwater and core cooling.

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)**

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**82** **ID: 1737141** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- LOCA requiring a plant trip.
- Pressurizer Level cannot be maintained without HPI.
- OP-TM-EOP-006 LOCA COOLDOWN is in progress.

(1) Which Reactor Coolant Pumps must get secured?
(2) What is the OP-TM-EOP-006 basis for securing them?

A. (1) RC-P-1A and RC-P-1B
(2) Minimize the RCS inventory loss due to the LOCA.

B. (1) RC-P-1A and RC-P-1B
(2) The subsequent cooldown could result in "core lift".

C. (1) RC-P-1C and RC-P-1D
(2) Minimize the RCS inventory loss due to the LOCA.

D. (1) RC-P-1C and RC-P-1D
(2) The subsequent cooldown could result in "core lift".

Answer: D

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) In OP-TM-EOP-006, if all four Reactor Coolant Pumps are running, two of them get secured; (2) The basis for this to ensure no more than three Reactor Coolant Pumps are running when Tcold drops below 407F; (3) With four Reactor Coolant Pumps the differential pressure may result in core lift; (4) OP-TM-EOP-006 directs RC-P-1C and RC-P-1D to be shutdown.

| | | |
|---|---|---|
| A. | (1) RC-P-1A and RC-P-1B (2) Minimize the RCS inventory lost due to the LOCA.(1) RC-P-1A and RC-P-1B (2) Minimize the RCS inventory lost due to the LOCA. | INCORRECT: Plausible but incorrect because the step indicates if RC-P-1A and RC-P-1B are running, THEN shutdown RC-P-1C and RC-P-1D. In addition the basis is to prevent core lift. |
| B. | (1) RC-P-1A and RC-P-1B (2) Lower the RCS flow and differential pressure across the fuel bundles. | INCORRECT: Plausible but incorrect because the step indicates if RC-P-1A and RC-P-1B are running, THEN shutdown RC-P-1C and RC-P-1D. |
| C. | (1) RC-P-1C and RC-P-1D (2) Minimize the RCS inventory lost due to the LOCA. | INCORRECT: Plausible because this action could lower the rate of RCS inventory loss on some LOCA. Incorrect because the Reactor Coolant Pumps are secured due to core lift concerns. |
| D. | (1) RC-P-1C and RC-P-1D (2) The subsequent cooldown could result in "core lift". | Correct Answer: See above. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 2 |
| | K/A # | BW/E08 | 2.4.18 |
| | Importance Rating | | 4.0 |

K/A: LOCA Cooldown: Knowledge of the specific bases for EOPs.

Proposed Question: Question #82

Technical Reference(s): OP-TM-EOP-006, Rev 12

OP-TM-EOP-0061, Rev 7

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP006-PCO-4

Question Source: Bank #

Modified Bank #

New X

# EXAMINATION ANSWER KEY
16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question History: N/A | | Last NRC Exam: N/A | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | b.5 | |

Comments:
K/A Match: This question matches the K/A because the examinee must know the reason for securing two Reactor Coolant Pumps on a LOCA Cooldown.

SRO ONLY: The question is SRO only because the examinee must assess plant conditions and know the content of OP-TM-EOP-006 in order to select a required course of action. In addition, the examinee must know the basis for the action selected. This procedure is not one of the major EOPs, it is a supplementary EOP directed from within the major EOP.

**83** **ID: 1737631** **Points: 1.00**

The specific activity of the primary and secondary coolant are as follows:

| | Two days ago at 1200 | Yesterday at 1200 | Today at 1200 |
|---|---|---|---|
| RCS Dose Equivalent I-131 | 0.50 microcuries/gram | 0.53 microcuries/gram | 0.52 microcuries/gram |
| OTSG Dose Equivalent I-131 | 0.05 microcuries/gram | 0.053 microcuries/gram | 0.052 microcuries/gram |

The ____(1)_____ is out of specification, and the basis of the specification is to maintain dose within the limits from ____(2)_____.

A. (1) RCS
(2) a tube rupture accident

B. (1) OTSG
(2) a steam line break accident

C. (1) RCS
(2) baseline OTSG tube leakage

D. (1) OTSG
(2) normal condenser offgas

Answer: A

**Answer Explanation**

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) The limit for I-131 in the RCS is 0.35 microcuries/gram; (2) The limit for I-131 in the secondary fluid is .10 microcuries/gram, which is not exceeded in this question; (3) The basis for the RCS limit is that if activity levels are kept below this level then analysis shows that the calculated doses are within acceptable limits on a Steam Line Break (SLB) and Steam Generator Tube Rupture (SGTR) accident (T.S. 3.1.4.1 basis)

| | | |
|---|---|---|
| A. | (1) RCS<br>(2) dose from a tube rupture accident | CORRECT ANSWER: See above. |
| B. | (1) OTSG<br>(2) dose from a steam line break accident | INCORRECT: Plausible if the examinee is not familiar with the secondary chemistry limits. In addition, the examinee could believe the secondary is the concern due to having one less barrier than the RCS to atmosphere. |
| C. | (1) RCS<br>(2) dose from baseline OTSG tube leakage | INCORRECT: Plausible because the RCS is violating the tech spec chemistry limits. Incorrect because the basis is not dose from a baseline OTSG tube leak. That is the basis for violating the secondary chemistry limits. |
| D. | (1) OTSG<br>(2) dose from normal condenser offgas | INCORRECT: Plausible if the examinee is not familiar with the secondary chemistry limits. In addition, dose from the normal condenser offgas could be a concern. Incorrect because the secondary chemistry is within limits. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | | 1 |
| Group # | | 2 |
| K/A # | 076 | 2.2.38 |
| Importance Rating | 3.6 | 4.5 |

K/A: High Reactor Coolant Activity: Knowledge of conditions and limitations in the facility license.

Proposed Question: Question #83

Technical Reference(s): T.S. 3.1.4
T.S. 3.1.13

Proposed References to be provided to applicants during examination: None

Learning Objective: 220-GLO-14

Question Source: Bank # 718419
Modified Bank #
New

Question History: System Exam 12 Last NRC Exam: 08-01

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | b.2 | |

Comments:

KA Match: This question matches the KA because the examinee must know the basis for the primary chemistry limits.

SRO ONLY: This question is SRO only because the examinee must identify the out of specification chemistry reading, and understand the basis behind the limit.

Correct answer. TS 3.1.4 states "With the specific activity of the primary coolant greater than 0.35 microcurie/gram DOSE I EQUIVALENT 1-131 for more than 48 hours*' during one continuous time interval or exceeding the limit line shown on Figure 3.1-24 be in at least HOT SHUTDOWN within 6 hours. Power operation may continue when DOSE EQUIVALENT 1-131 is below 0.35 microcurie/gram."

Basis "The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will be well within the Part 100 limit following a steam generator tube rupture accident or steam line break accident with postulated accident induced steam generator tube leakage in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM."

Plausible since this is the specific activity limit for the Secondary Coolant and it is based on a steam line rupture with tube leakage; however the specific activity limit is 0.35 microcurie/gram for the Primary Coolant and the primary-to-secondary steam generator leakage rate of 1.0 GPM is assumed.

Plausible since the limit and basis is correct; however it is correct for the Primary Coolant.

3.13 SECONDARY COOLANT SYSTEM ACTIVITY

3.13.1 The specific activity of the secondary coolant system shall be ≤ 0.10 - Ci/gram DOSE EQUIVALENT I-131.

3.13.2 With the specific activity of the secondary coolant system > 0.10-Ci/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

Bases

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary-to-secondary tube leak in the steam generator of the affected steam line.

Plausible since the limit is correct for the Secondary Coolant; however the basis is for a coincident 1.0 GPM primary-to-secondary tube leak.

**84** **ID: 1699987** **Points: 1.00**

The following plant conditions exist:

- A small break LOCA is in progress.
- The Reactor tripped, and ESAS actuated on low RCS pressure.
- A loss of off-site power (LOOP) occurred.
- Sub cooling margin was lost and is currently –1°F on the PT plot.
- Incore temperatures are rising slowly.
- RCITS shows a large steam void in each RCS hot leg.
- Current cooldown rate is 45°F / hr.

Which ONE of the following describes the correct actions the crew must take to ensure adequate core cooling?

A. OTSG level must be maintained above 50%, with OTSG saturation temperature from 40°F to 60°F below RCS cold leg temperature to promote boiler-condenser cooling. No additional action is required due to the steam void.

B. OTSG level must be maintained above 50%, with OTSG saturation temperature from 40°F to 60°F below RCS cold leg temperature to allow single-phase natural circulation. The steam void must be vented from the hot leg.

C. OTSG level must be maintained 75% - 85%, with OTSG pressure at least 100 psig below RCS pressure to promote boiler-condenser cooling. No additional action is required due to the steam void.

D. OTSG level must be maintained 75% - 85%, with OTSG pressure at least 100 psig below RCS pressure to allow single-phase natural circulation. The steam void must be vented from the hot leg.

Answer: C

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) The controlling procedure is OP-TM-EOP-002, LOSS OF 25F SUBCOOLING MARGIN; (2) In addition to implementing actions OP-TM-EOP-002, the operating crew will be performing OP-TM-EOP-010, EMERGENCY PROCEDURE RULES, GUIDES, AND GRAPHS, in which the entry criteria for Rule 4, FEEDWATER CONTROL, and Guide 6, OTSG PRESSURE CONTROL have ben met, which provide governance on how to promote heat transfer in loss of 25F subcooling event; (3) In addition steps are taken to remove non-condensible gasses from the RCS (if there were symptoms of them being present), NOT steam.

| | | |
|---|---|---|
| A. | OTSG level must be maintained above 50%, with OTSG saturation temperature from 40°F to 60°F below RCS cold leg temperature to promote boiler-condenser cooling. No additional action is required due to the steam void. | INCORRECT ANSWER: Plausible because maintaining greater than 50% in the operating range is the level band if reactor coolant pumps are secured. Maintaining OTSG saturation temperature in the 40-60F band below RCS cold leg temperature is the band if the cooldown rate would be less than 40 F/hr. OP-TM-EOP-004 also addresses how to address a steam void of non-condensable gasses in the RCS. In this case, this would not be necessary due to the cooldown rate of > 40F/hr being present. |
| B. | OTSG level must be maintained above 50%, with OTSG saturation temperature from 40°F to 60°F below RCS cold leg temperature to allow single-phase natural circulation. The steam void must be vented from the hot leg. | INCORRECT ANSWER: Plausible because maintaining greater than 50% in the operating range is the level band if reactor coolant pumps are secured. Maintaining OTSG saturation temperature in the 40-60F band below RCS cold leg temperature is the band if the cooldown rate would be less than 40 F/hr. OP-TM-EOP-004 also addresses how to address a steam void of non-condensable gasses in the RCS. In this case, this would not be necessary due to the cooldown rate of > 40F/hr being present. |
| C. | OTSG level must be maintained 75% - 85%, with OTSG pressure at least 100 psig below RCS pressure to promote boiler-condenser cooling. No additional action is required due to the steam void. | CORRECT ANSWER. Both actions are required by OP-TM-EOP-010 (Rule 4 and Guide 6). Steps to vent are in OP-TM-EOP-004, which was not entered in this question. In addition, venting is only performed if it is believed non-condensable gasses are present in the RCS, not steam. |
| D. | OTSG level must be maintained 75% - 85%, with OTSG pressure at least 100 psig below RCS pressure to allow single-phase natural circulation. The steam void must be vented from the hot leg. | INCORRECT ANSWER: Plausible because both actions are required by OP-TM-EOP-010 (Rule 4 and Guide 6). Steps to vent are in OP-TM-EOP-004, which was not entered in this question. In addition, venting is only performed if it is believed non-condensable gasses are present in the RCS, not steam. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| | Group # | | 2 |
| | K/A # | E03 | EA2.1 |
| | Importance Rating | | 4.0 |

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

K/A: Inadequate Subcooling Margin: Ability to determine and interpret the following as the apply to the (Inadequate Subcooling Margin): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

Proposed Question: Question #84

Technical Reference(s): OP-TM-EOP-002

OP-TM-EOP-010, Rev 19

Proposed References to be provided to applicants during examination: None

Learning Objective: N-TM-TQ-104-EOP-DBIG-APCO -1

Question Source: Bank # 862875

Modified Bank #

New

Question History: Comp 1 Exam Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 b.5

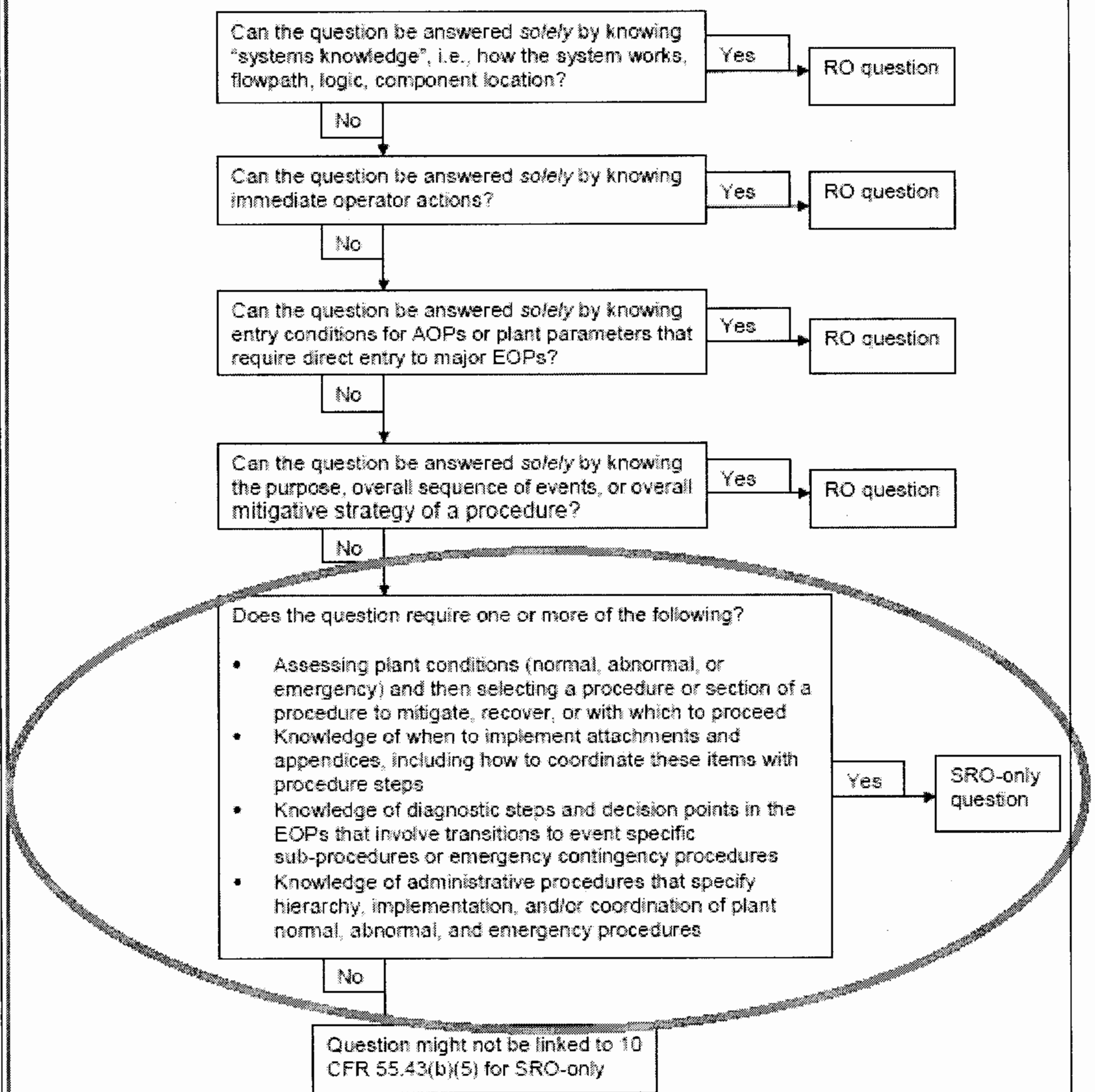
Comments:

K/A Match: This matches the K/A because the examinee will have to know which procedures to use to promote cooling in the RCS. Although not explicitly stated, by asking steps in Rule 4 and Guide 6, the examinee will have selected which procedure is appropriate. In addition, to rule out the two of the distractors, the examinee will NOT have selected to choose another plausible EOP network path into OP-TM-EOP-004.

High Cog: This question is high cog because the examinee must analyze the given set of parameters and choose the correct procedure and steps in which to execute.

SRO Only: This question is an SRO only question because it the examinee must assess plant conditions and then select procedure steps to mitigate the casualty. This question tests coordination between how the operating crew will feed and steam the OTSGs.

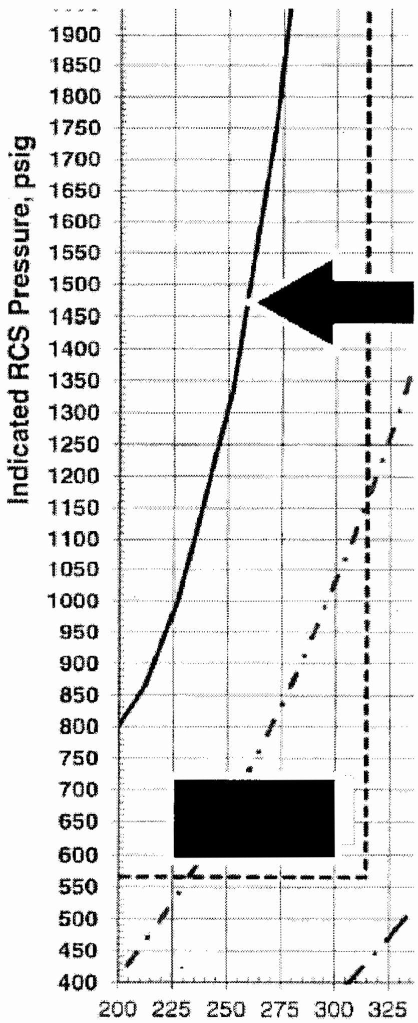
**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**85** **ID: 1720678** **Points: 1.00**




<<

REFERENCE PROVIDED

Plant conditions:

- Cooldown in progress in accordance with OP-TM-EOP-006, LOCA COOLDOWN.

EVENT:

- Due to a plant upset, RCS temperature is approaching the curve on Figure 1, RCS PRESSURE-TEMPERATURE LIMITS.

The operating crew must ensure open the __(1)__ to prevent exceeding the__(2)__, in accordance with Guide 23, RCS PRESSURE AND TEMPERATURE LIMITS.

A. (1) PORV
(2) TS NDT Curve

B. (1) PORV
(2) PZR SURGE LIMIT Curve

C. (1) Spray Valve
(2) TS NDT Curve

D. (1) Spray Valve
(2) PZR SURGE LIMIT Curve

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) During and LOCA Cooldown, OP-TM-EOP-010, Guide 23 and Figure 1 and 1a are used to maintain RCS pressure and temperature within limits; (2) The line that the arrow is point to is the TS NDT line, which is a composite curve to Tech Spec Figures 3.1-1 and 3.1-2 based on the reactor vessel individual components and welds; (3) The PORV is specified for RCS pressure control to move RCS pressure away from the limits described by the curve because RCS pressure reduction is the most effective method from moving away from the curve.

| | | |
|---|---|---|
| A. | (1) Open the PORV<br>(2) TS NDT Curve | Correct Answer - Guide 23, Step 1 : If RCS pressure is approaching "TS NDT limit", then OPEN RC-RV-2 (PORV). |
| B. | (1) Open the PORV<br>(2) PZR SURGE LIMIT Curve | Plausible because this curve is in the same general area, but on FIGURE 1A, LOW RANGE RCS PRESSURE-TEMPERATURE LIMITS. The action for exceeding that curve is to lower RCS pressure or raise RCS temperature. Incorrect because the curve referenced in the picture is the TS NDT Curve. |
| C. | (1) Open the Spray Valve<br>(2) TS NDT Curve | Plausible because opening the Spray Valve will lower pressure away from the TS NDT Curve. Incorrect because GUIDE 23 directs use of the PORV to lower pressure. |
| D. | (1) Open the Spray Valve<br>(2) PZR SURGE LIMIT Curve | Plausible because if the examinee believed that the PZR SURGE LIMIT Curve was being reference then opening the spray valve would be an acceptable method for lowering pressure. Incorrect because the TS NDT Curve is being referenced. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | | 1 |
| Group # | | 2 |
| K/A # | BW/E13 | 2.2.42 |
| Importance Rating | | 4.6 |

K/A: EOP Rules and Guides: Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question: Question #85

Technical Reference(s): OP-TM-EOP-006, Rev 12 T.S Figure 3.1-1
OP-TM-EOP-0061, Rev 7 OP-TM-EOP-010, Rev 19

Proposed References to be provided to applicants during examination: OP-TM-EOP-010, Figure 1

Learning Objective: EOP006-PCO-4

Question Source: Bank #
Modified Bank #
New X

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question History: N/A | | Last NRC Exam: N/A | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | b.2 | |

Comments:

KA Match: This question matches the KA because the examinee will have to recognize that RCS pressure-temperature is approaching the TS NDT Curve, which would be a violation of Technical Specifications if violated.

High Cog: This question is high cog because the examinee will have to analyze the situation and choose the correct course of action.

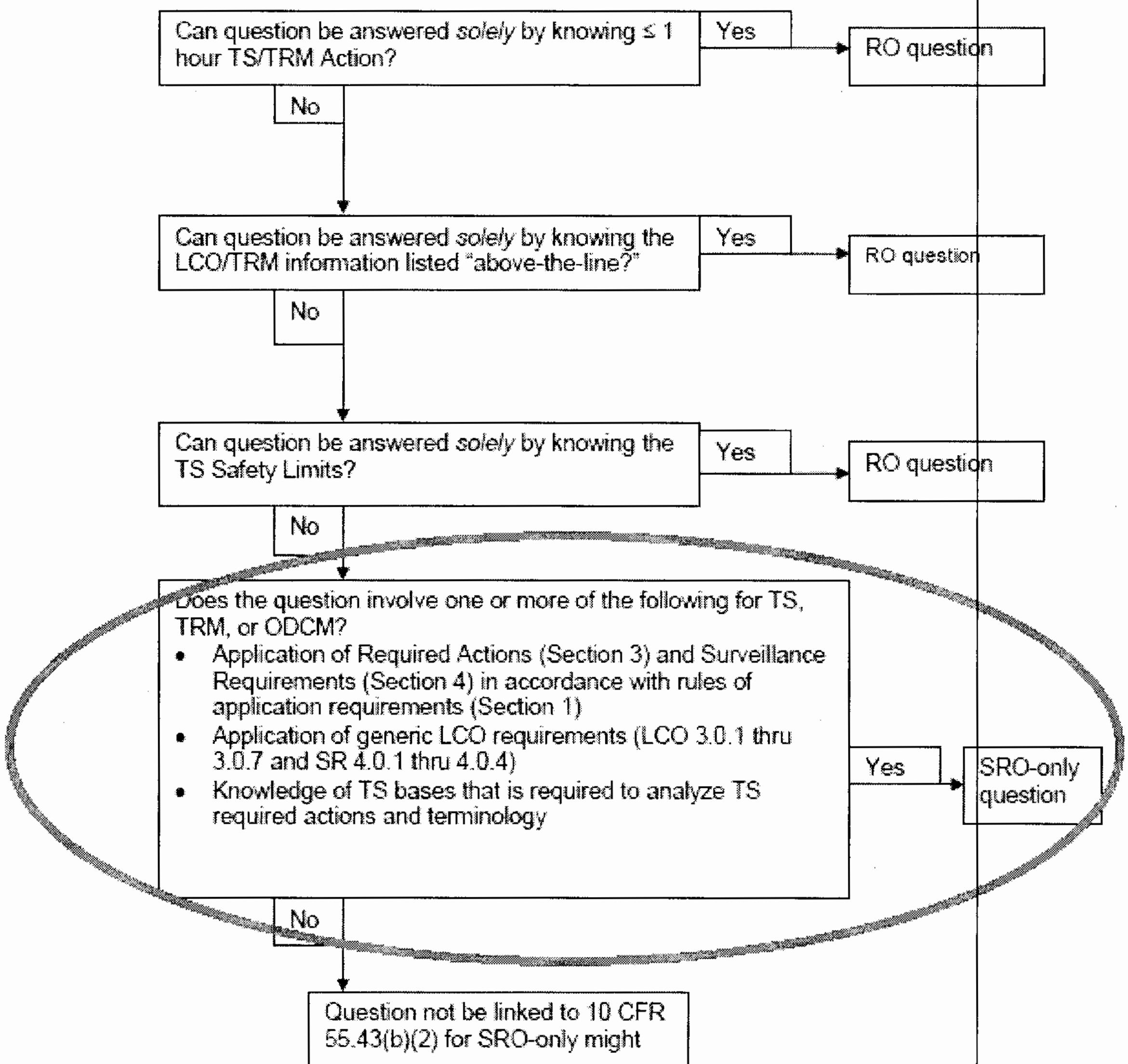
SRO Only: This question is SRO only because it involves recognizing that a TS NDT limit is going to be violated and choosing the correct action to prevent violation.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

ES-401 | 5 | Attachment 2

**Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**86** **ID: 1740783** **Points: 1.00**

Plant Conditions:

- Refueling Operations in progress.
- DH Loop A in operation.

Event:

- DH-P-1A trips on overcurrent.
- Incore and RCS temperatures have risen from 134°F to 145°F and are currently steady.

Based on these conditions, identify the ONE selection below that describes:

(1) Whether a Reactor Operating Mode change has occurred.
(2) Procedure to be used to respond to the event.

A. (1) A Reactor Operating Mode change has occurred.
(2) EOP-030, Loss of Decay Heat Removal.

B. (1) A Reactor Operating Mode change has NOT occurred.
(2) EOP-030, Loss of Decay Heat Removal.

C. (1) A Reactor Operating Mode change has occurred.
(2) OP-TM-212-901, Emergency DHR Operations.

D. (1) A Reactor Operating Mode change has NOT occurred.
(2) OP-TM-212-901, Emergency DHR Operations.

Answer: A

**Answer Explanation**

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | |
|---|---|---|
| << | | |
| Explanation: To answer this question correctly, the examinee must know: (1) When DH-P-1A trips, since the fuel is not fully unloaded, the RCS temperature could rise; (2) Init ally, the crew would enter the alarm response OP-TM-MAP-B0102, 4KV ES MOTOR TRIP, which would direct the crew to init ate OP-TM-211-112, SHIFTING DH TRAIN B FROM DHR STANDBY TO DHR OPERATION; (3) While the crew may be working on shif ing DHR to train B, when incore temperature has risen more than 10F, OP-TM-EOP-030, LOSS OF DECAY HEAT REMOVAL is entered; (4) The mode changes from REFUELING SHUTDOWN to COLD SHUTDOWN when RCS Temperature rises above 140°F. | | |
| A. | (1) A Reactor Operating Mode change has occurred. (2) EOP-030, Loss of Decay Heat Removal. | Correct Answer: See above. |
| B. | (1) A Reactor Operating Mode change has NOT occurred. (2) EOP-030, Loss of Decay Heat Removal. | Incorrect Answer: Plausible if the examinee does not know that RCS temperature rising about 140°F. |
| C. | (1) A Reactor Operating Mode change has occurred. (2) OP-TM-212-901, Emergency DHR Operations. | Incorrect Answer: Plausible because the OP-TM-212-901 would be entered if an emergency condit on existed in accordance with OP-TM-MAP-B0102. Incorrect because RCS temperature exceeded 140°F, OP-TM-EOP-030 must be entered. |
| D. | (1) A Reactor Operating Mode change has NOT occurred. (2) OP-TM-212-901, Emergency DHR Operations. | Incorrect Answer: Plausible if the examinee does not know that RCS temperature rising about 140°F. In addit on, Plausible because the OP-TM-212-901 would be entered if an emergency condition existed in accordance with OP-TM-MAP-B0102. Incorrect because RCS temperature exceeded 140°F, OP-TM-EOP-030 must be entered. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 006 | 2.4.9 |
| | Importance Rating | | 4.2 |

K/A: Emergency Core Cooling: Knowledge of lower power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

Proposed Question: Question #86

Technical Reference(s): OP-TM-MAP-B0102, Rev 2 OP-TM-EOP-030, Rev 10

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-030-PCO-2

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question Source: | Bank # | 371862 | |
| | Modified Bank # | | |
| | New | | |
| Question History: | Comp 2 | Last NRC Exam: | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | X |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | b.5 | |

Comments:

KA Match: This question matches the KA because the examinee must know the mitigation strategy for a loss of decay heat removal during a shutdown situation.

High Cog: This question is high cog because the examinee must analyze conditions in the stem, and choose the correct course of action.

SRO only: This question is SRO only because the examinee must know what the tech spec definitions for REFUELING SHUTDOWN and COLD SHUTDOWN, and they apply to the stem of the question.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**87** **ID: 1720770** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- A worker reported a fire in the Relay Room and evacuated the area.
- The operating crew has entered OP-TM-EOP-020, COOLDOWN OUTSIDE OF CONTROL ROOM.
- The crew has tripped both Main Feedwater Pumps, all running Condensate Booster Pumps, and all running Condensate Pumps.

(1) What attachment will the Primary Safe Shutdown NLO perform?
(2) What event will this attachment prevent or terminate?

A. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Terminate uncontrolled HPI due to a spurious "A" train ES actuation.

B. (1) Attachment 5, "Preventing Spurious Operation of MOV's"
(2) Prevent an overcooling event by preventing MS-V-2A/B (Isolations to EF-P-1, TBV's and ADVs) from spuriously opening.

C. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent the Reactor Coolant Pumps from spuriously starting.

D. (1) Attachment 13, "Tripping RCPs Locally"
(2) Prevent Reactor Coolant pumps from operating with no seal injection.

Answer: A

**Answer Explanation**

# EXAMINATION ANSWER KEY

## 16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) OP-TM-EOP-020, COOLDOWN FROM OUTSIDE OF CONTROL ROOM is entered when there is a fire in the control room or relay room that has a potential to cause damage to safe shutdown required equipment; (2) Once the determination is made that the control room must be evacuated, this starts a sequence in which actions are taken to prevent or terminate undesirable actions which may be caused by the fire; (3) With the announcement at the end of the IMA's, control room operators and non-licensed operators (NLOs) are directed to perform actions within attachments to mitigate possible effects of the fire; (4) The NLOs respond in accordance with OP-TM-EOP-020, and OS-24, CONDUCT OF OPERATIONS DURING ABNORMAL AND EMERGENCY EVENTS, which allows to them to take actions prior to having CRS concurrence (Attachment F, of OS-24), which includes performing Attachment 5 and Attachment 13 of OP-TM-EOP-020; (5) The Primary Safe Shutdown NLO will perform Attachment 5, which among other things, will terminate uncontrolled HPI due to a spurious "A" train ES actuation by opening the breakers and locally closing MU-V-16A and B.

| | | |
|---|---|---|
| A. | (1) Attachment 5, "Preventing Spurious Operation of MOV's"<br>(2) Terminate uncontrolled HPI due to a spurious "A" train ES actuation. | Correct Answer - The Primary Safe Shutdown NLO will perform attachment 5. |
| B. | (1) Attachment 5, "Preventing Spurious Operation of MOV's"<br>(2) Prevent an overcooling event by preventing MS-V-2A/B (Isolations to EF-P-1, TBV's and ADVs) from spuriously opening. | Incorrect Answer - Plausible because the MS-V-8's (Isolation valves to the Turbine Bypass Valves) are closed to prevent an overcooling event due the turbine bypass valves failing midscale on some fire events. The MS-V-2's could also be used to isolated the turbine bypass valves, but closing the MS-V-2 would also isolate a steam supply path to EF-P-1 (Steam Driven Emergency Feedwater Pump) and an MS-V-4 (Atmospheric Dump Valve). Attachment 5 does open the breaker for MS-V-2A/B, but that is to maintain the valve open for use of EF-P-1 and the MS-V-4's. |
| C. | (1) Attachment 13, "Tripping RCPs Locally"<br>(2) Prevent the Reactor Coolant Pumps from spuriously starting. | Incorrect Answer - Plausible because Attachment 13 is performed by the Secondary Safe Shutdown NLO. Preventing Reactor Coolant Pumps from spuriously starting is one basis for performing Attachment 13. |
| D. | (1) Attachment 13, "Tripping RCPs Locally"<br>(2) Prevent Reactor Coolant pumps from operating with no seal injection. | Incorrect Answer - Plausible because Attachment 13 is performed by the Secondary Safe Shutdown NLO. Preventing Reactor Coolant Pumps from operating without seal injection is one basis for performing Attachment 13. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 013 | 2.4.35 |
| | Importance Rating | | 4.0 |

K/A: Engineered Safety Features Actuation: Knowledge of local auxiliary operator tasks during an emergency and their resultant operational effects.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

Proposed Question: Question #87

Technical Reference(s): OP-TM-EOP-020, Rev 22 OS-24, Rev 28

OP-TM-EOP-0201, Rev 15

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP020-PCO-1

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 b.5

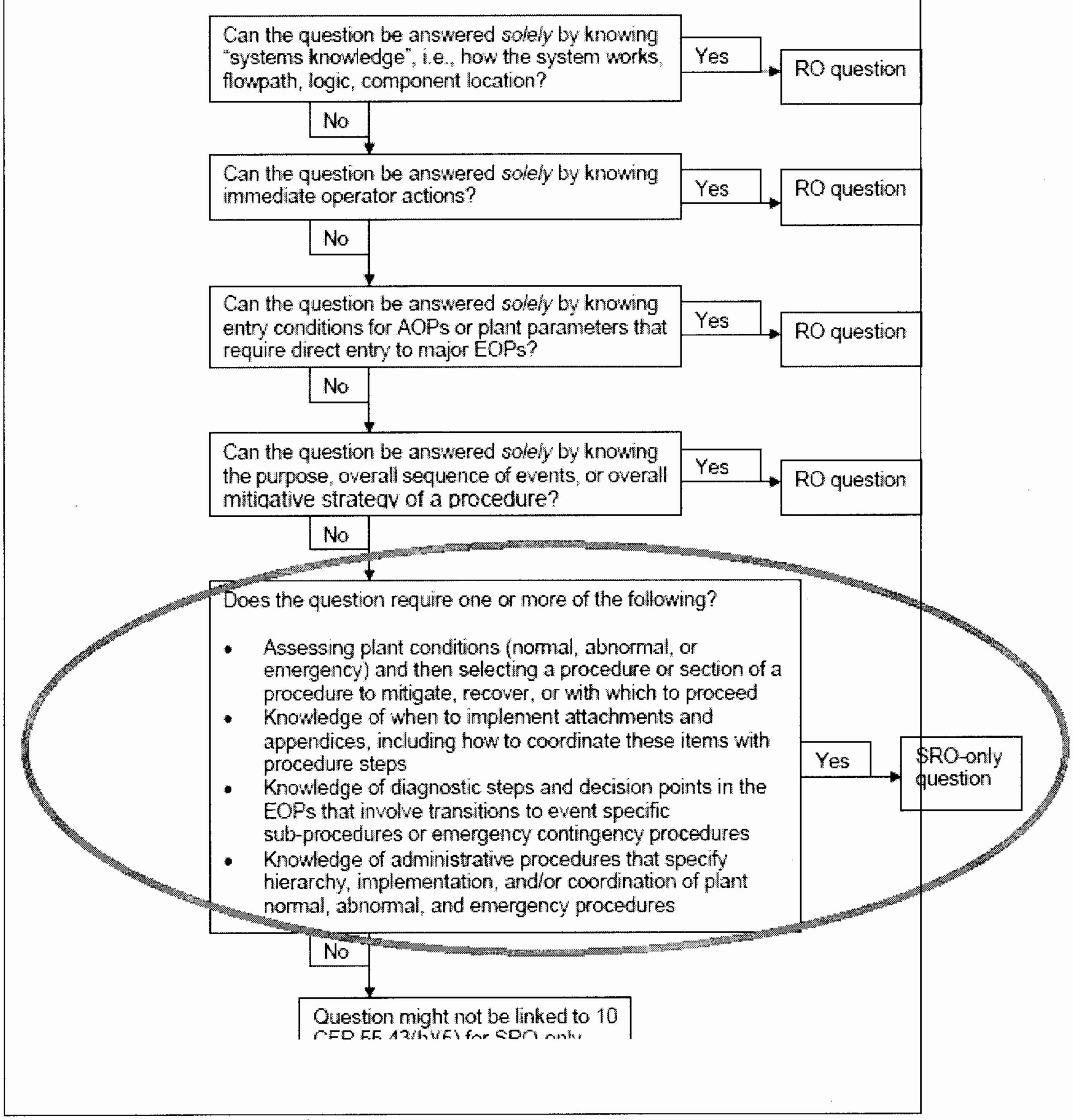
Comments:

KA Match: This question matches the KA because the examinee will have to know that a fire in the relay room could cause and unwanted HPI actuation, but that the Attachment 5 will close MU-V-16A and B which would mitigate the consequence of such event.

High Cog: This question is high cog because the examinee will have to assess plant conditions to determine that a control room evacuation is required and that the remote shutdown sequence has been directed. Step 2.8 is a stopping point in which the operating crew would wait at until conditions in the control room have deteriorated enough that crew must evacuate. This is denoted by the tripping of the feedwater, condensate booster, and condensate pumps.

SRO Only: This question is SRO only because the examinee will have to assess that the stem in a way that determines that the remote shutdown sequence has begun, and understand which attachment the NLOs have to perform.

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**88** **ID: 1720789** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

EVENT:

- Main and Emergency Feedwater are both lost.
- HPI/PORV cooling is in progress.

POST EVENT:

- Both Main and Emergency Feedwater are restored.
- One Reactor Coolant Pump is running in each loop.
- Both OTSG's are intact and dry.
- PPC Points C4015 and C4016 (OTSG tube-to-shell differential temperatures) are at +70F.

In accordance with Guide 13, FEEDING A DRY OR DEPRESSURIZED OTSG, __(1)__ is the preferred feed source because it will __(2)__.

A. (1) Main Feedwater
(2) cool down the OTSG tubes faster

B. (1) Main Feedwater
(2) minimize the tensile loading on the OTSG tubes

C. (1) Emergency Feedwater
(2) cool down the OTSG tubes faster

D. (1) Emergency Feedwater
(2) minimize the tensile loading on the OTSG tubes

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) The stem sets up a scenario in which the plant is experiencing a lack of primary to secondary heat transfer (i.e. Reactor tripped, no feedwater available) which heats up the RCS, and thus the OTSG tubes heat; (2) While the tubes heat up, the OTSG shell temperature should remain constant or slightly cooldown due to the lack of feedwater; (3) This causes a compressive stress on the OTSG tubes (which is a positive TSDT); (4) This is also indicated by the (+) on computer points C4015 and C4016; (5) Being above the (+) limit of TSDT means there is a compressive stress on the OTSGs; (6) The actions for a (+) TSDT in Guide 13 with a reactor coolant pump on is to feed the OTSG with EFW; (7) In accordance with the Guide 13 basis document, EFW is chosen because it sprays directly on the OTSG tubes, and will lower the tube temperature immediately, thus lowering the TSDT and compressive stresses; (8) Since EFW sprays directly on the tubes it will cool down the tubes faster than MFW.

| | | |
|---|---|---|
| A. | (1) Main Feedwater<br>(2) cool down the OTSG tubes faster | Incorrect - Plausible because in most cases when a reactor coolant pump is on, MFW is the preferred source of feedwater. This would be the case if the TSDT was negative. In addition, MFW would not cool down the OTSG tubes faster. |
| B. | (1) Main Feedwater<br>(2) minimize the tensile loading on the OTSG tubes | Incorrect - Plausible because in some cases Main Feedwater would minimize the tensile loading on the OTSG tubes. Incorrect be cause EFW is the preferred source of feedwater. |
| C. | (1) Emergency Feedwater<br>(2) cool down the OTSG tubes faster | Correct Answer: See above. |
| D. | (1) Emergency Feedwater<br>(2) minimize the tensile load on the OTSG tube | Incorrect Answer: Plausible because EFW is the preferred source. Incorrect because the stress currently exhibited on the OTSG tubes is compressive. Spraying EFW on the tubes would minimize the compressive loading on the tubes, not the tensile loading. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 061 | 2.4.18 |
| | Importance Rating | | 4.0 |

K/A: Auxiliary/Emergency Feedwater: Knowledge of the specific basis for EOPs.

Proposed Question: Question #88

Technical Reference(s): OP-TM-EOP-010, Guide 13, Rev 19

OP-TM-EOP-0101, Guide 13, Rev 9

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP010-PCO-4

Question Source: Bank #

| | | | | |
|---|---|---|---|---|
| | Modified Bank # | | | |
| | New | X | | |
| Question History: | N/A | | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | | X | |
| 10 CFR Part 55 Content: | 55.41 | | | |
| | 55.43 | b.5 | | |

Comments: (KA Match, why high cog, why SRO only)

KA Match: This question matches the KA because the examinee must know the basis for choosing EFW to feed the OTSG.

High Cog: The question is high cog because the examinee must determine which step of Guide 13 which applies from the conditions in the stem.

SRO Only: This question is SRO only because it requires the examinee to assess plant conditions (Restoring from lack of heat transfer, MFW and EFW available, RCP running, positive TSDT) and select a course of action (feed with EFW because a RCP is on and TSDT is positive). Guide 13 is not a major EOP, but is directed from other major EOPs.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**89** **ID: 1737969** **Points: 1.00**

Plant conditions:

- 100% power with ICS in full auto.
- Backup Instrument Air Compressor IA-P-2A is tagged out of service for maintenance.
- Main Vacuum Pump VA-P-1A is operating.

Sequence of events:

| Time | Event |
| --- | --- |
| 0100 | Local fire (now extinguished) renders Service Air Compressor's SA-P-1A and SA- P-1B inoperable. |
| 0200 | MAP PLB-1-6 IA-P-4 IA-Q-2 TROUBLE actuates. |
| 0210 | Instrument Air Compressor IA-P-1A trips on motor overcurrent. |
| 0220 | Instrument Air Compressor IA-P-1B trips on motor overcurrent. |
| 0230 | The following valid alarms actuate:<br>PLB-1-7 Instrument Air Press Low Turbine Area.<br>PLB-1-8 Station Service Air Press Low.<br>PLB-2-7 Instrument Air Press Low Aux Bldg Area. |
| 0240 | Current Conditions:<br>• All Instrument and Service Air System pressure indications are now at 75 psig, reducing at 1 psi per minute.<br>• Main Condenser vacuum is 27 inches Hg, reducing at 0.2 inches Hg per minute.<br>• Instrument Air Compressor IA-P-4 filter and dryer d/p at 22 psid |

What action is required for these conditions that will prevent entry into Technical Specification LCO 3.4, Decay Heat Removal Capability?

A. Trip the reactor.

B. Take manual control of MU-V-20.

C. Open VA-V-5A vacuum pump suction valve.

D. Ensure open IA-Q-2 bypass valve IA-V-2133 and open IA-V-2124 (Pre-filter Bypass Valve).

Answer: D

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) When instrument air becomes less than 80 psig, OP-TM-AOP-028, LOSS OF INSTRUMENT AIR, is entered; (2) In OP-TM-AOP-028, there are various actions taken for the lowering levels for air pressures or high level of differential pressures; (3) The parameter exceeded for this scenario is high D/P for IA-P-4 filter and dryer ( > 20 psid); (4) The Turbine Bypass Valves, (TBVs) MS-V-3A-F usually receive air from the instrument air header, and have backup air from the 'A' Backup Instrument Air Compressor, IA-P-2A; (5) Since IA-P-2A is OOS in the stem, and with the various events occurance to the other compressors, when the IA-P-4 filter and dryer DP becomes > 22 psid, the MS-V-3's become inoperable; (6) One of the purposes of the TBVs is to remove decay heat when the reactor is shutdown, and 4 of 6 must be operable per technical specification 3.4.1.1.

| | | |
|---|---|---|
| A. | Trip the reactor. | INCORRECT: Manual reactor trip at 60 psig is a required action to in order to avoid potential loss of control or erratic behavior of air operated components at significant power levels. Distracter is plausible because this answer represents a valid IAAT action at less than 60 psig, and it does result in reduction of core heat production by decay of fission products. In addition, the examinee could believe that tripping the reactor is an option to not enter a tech spec, as some tech specs only apply when the reactor is critical. |
| B. | Take manual control of MU-V-20. | INCORRECT: The purpose of this action is to prevent loss of reactor coolant pump seal injection flow. Distracter is plausible because this answer represents a valid manual action included in AOP-028. In addition, MU-V-20 is a containment isolation valve, the examinee could believe that MU-V-20 fails open (it does not) which would represent a loss of containment and potentially an entry into the containment tech spec. |
| C. | Open VA-V-5A vacuum pump suction valve. | INCORRECT: Plausible because the examinee could believe that vacuum is required for the TBV's to be operable. Distracter is plausible because VA-V-5A/B/C all fail closed on loss of instrument air. |
| D. | Ensure open IA-Q-2 bypass valve IA-V-2133 and open IA-V-2124 (Pre-filter Bypass Valve). | Correct Answer: IA-P-4 is the last air compressor available that could provide air pressure to the turbine bypass valves. In addition to all of the other failures, if the dryer and filter were clogged, they could be bypassed to provide air to the air header. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 078 | A2.01 |
| | Importance Rating | | 2.9 |

K/A: Instrument Air: Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Air dryer and filter malfunctions

Proposed Question: Question #89

Technical Reference(s): OP-TM-AOP-028, Rev 9

Tech Spec 3.4.1.1

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-028-PCO-4

Question Source: Bank # 371312

Modified Bank #

New

Question History: Comp 1 Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 b.5

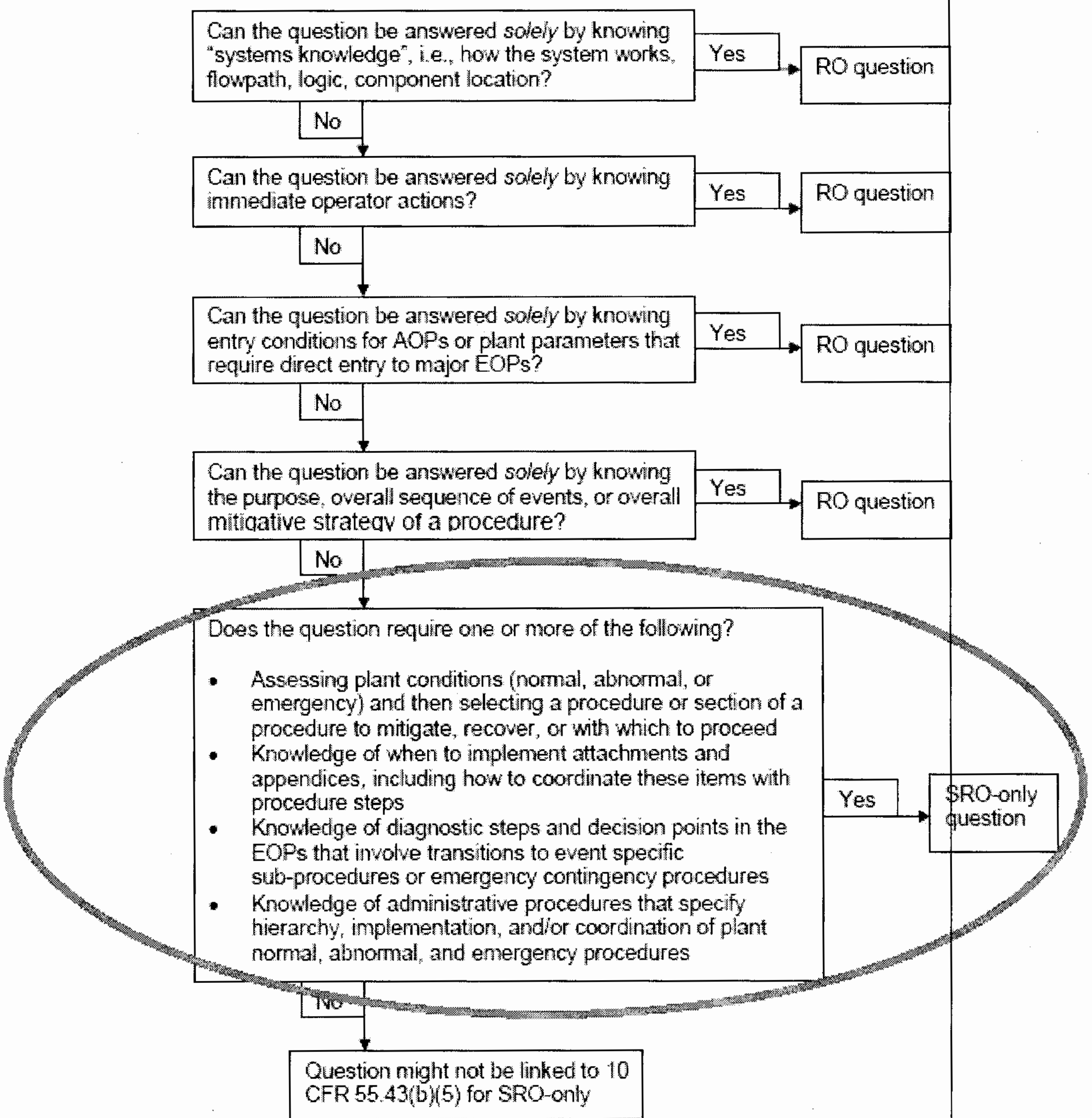
Comments:

KA Match: This question matches the KA because the examinee must identify which parameter is out of specification due to an instrument air failure and determine the procedure step which will remedy the situation.

High Cog: This question is high cog because the examinee analyze the current parameters and trends of the instrument air system and make a decision on the correct course of action to ensure the plant does not enter a tech spec LCO.

SRO Only: This is an SRO only question because the examinee must assess plant conditions then select the appropriate procedure to mitigate the malfunctions. In addition, the SRO must know the content of the procedure in which to evaluate the stems parameters with, and choose the correct course of actions based on the assessment.

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)**

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# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**90** **ID: 1737999** **Points: 1.00**

Plant Conditions:

- The plant is shut down in a Refueling Outage.
- Core off-load in progress.
- Containment Integrity is NOT being maintained.
- RB Purge is in progress IAW OP-TM-823-408 RB Purge - RB Doors and/or Equipment Hatch Open.

Event:

- The Fuel Handling SRO reports that a Fuel Assembly has been dropped and damaged in the Fuel Transfer Canal.
- MAP C-1-1 RADIATION LEVEL HI, alarms.
- RB Purge Exhaust Duct Monitor RM-A-9 is off-scale high.
- RB Purge Exhaust Duct Monitor RM-A-9 Hi-Hi is 3000 CPM.
- NI-11 counts have gone up and steadied out at 30 cps.

In accordance with OP-TM-MAP-C0101, all personnel are __(1)__ to be evacuated from the reactor building, and the crew must __(2)__ the RB purge is secured in accordance with OP-TM-244-911, CONTAINMENT CLOSURE.

A. (1) required
(2) ENSURE

B. (1) required
(2) VERIFY

C. (1) NOT required
(2) ENSURE

D. (1) NOT required
(2) VERIFY

Answer: A

**Answer Explanation**

Explanation: To answer this question correctly, the examinee must know: (1) During fuel handling, the containment is allowed to be open as long a reactor building purge is in service with the RB Purge Valve interlocks in defeat (in addition to other requirements for closing the reactor building which are not pertinent to the question); (2) When the fuel assembly is dropped in the fuel transfer canal gasses are released, which cause RM-A-9 to alarm, which would cause MAP C-1-1 alarm in the control room; (3) The alarm response for RM-A-9 directs the evacuation of ALL personnel from the containment building, and to initiate OP-TM-244-911, CONTAINMENT CLOSURE; (4) OP-TM-244-901 directs the closing of AH-V-1A, B, C, and D (RB purge valves) and to ensure containment closure devices outside of containment are set when an ES actuation has occured and a valve has failed to close; (5) The NI-11 counts going up and steadying out at 30 cps is to enhance the plausibility of a distractor and for a better match of the KA as there would be an automatic RB evacuation alarm at 40 cps.

| | | |
|---|---|---|
| A. | (1) required<br>(2) ENSURE | Correct Answer - See Above |
| B. | (1) required<br>(2) VERIFY | INCORRECT - Plausible because OP-TM-823-408 and 1101-3, CONTAINMENT INTEGRITY AND ACCESS LIMITS directs the purge interlocks to close the purge valves to be defeated. The examinee must know that since the interlocks are defeated that the purge will not secure. If the examinee believes that the purge should have been secured, they would believe that they are VERIFYING the purge is secured rather than actually ENSURING (by actually securing) the purge. |
| C. | (1) NOT required<br>(2) ENSURE | INCORRECT - Plausible because NI-11 never gets over 40 cps, which would initiate an automatic RB Evacuation alarm. Not going over 40 cps is plausible because the fuel assembly was dropped in the fuel transfer canal, and not the reactor vessel. |
| D. | (1) NOT required<br>(2) VERIFY | INCORRECT - Plausible because NI-11 never gets over 40 cps, which would initiate an automatic RB Evacuation alarm. Not going over 40 cps is plausible because the fuel assembly was dropped in the fuel transfer canal, and not the reactor vessel. In addition, plausible because OP-TM-823-408 and 1101-3, CONTAINMENT INTEGRITY AND ACCESS LIMITS directs the purge interlocks to close the purge valves to be defeated. The examinee must know that since the interlocks are defeated that the purge will not secure. If the examinee believes that the purge should have been secured, they would believe that they are VERIFYING the purge is secured rather than actually ENSURING (by actually securing) the purge. |

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 103 | A2.04 |
| | Importance Rating | | 3.6 |

K/A: Containment: Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use the procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Containment evacuation (including recognition of the alarm)

Proposed Question: Question #90

Technical Reference(s): 1101-3, Rev 93A — OP-TM-MAP-C0101, Rev 3
OP-TM-244-911, Rev 4 — OP-TM-823-408, Rev 12

Proposed References to be provided to applicants during examination: None

Learning Objective: 661-GLO-5

Question Source: Bank #
Modified Bank # 1110308
New

Question History: Sim Exam 9 Last NRC Exam: Unmodified on 10-02

Question Cognitive Level: Memory or Fundamental Knowledge
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41
55.43 b.5

Comments:

KA Match: This question matches the KA because the examinee must predict what the impact of a dropped fuel assembly will be on the operations going on in the reactor building during refueling. From those predictions, the examinee must know the content of the containment closure procedure (OP-TM-244-911) to know that the RB purge valves get closed.

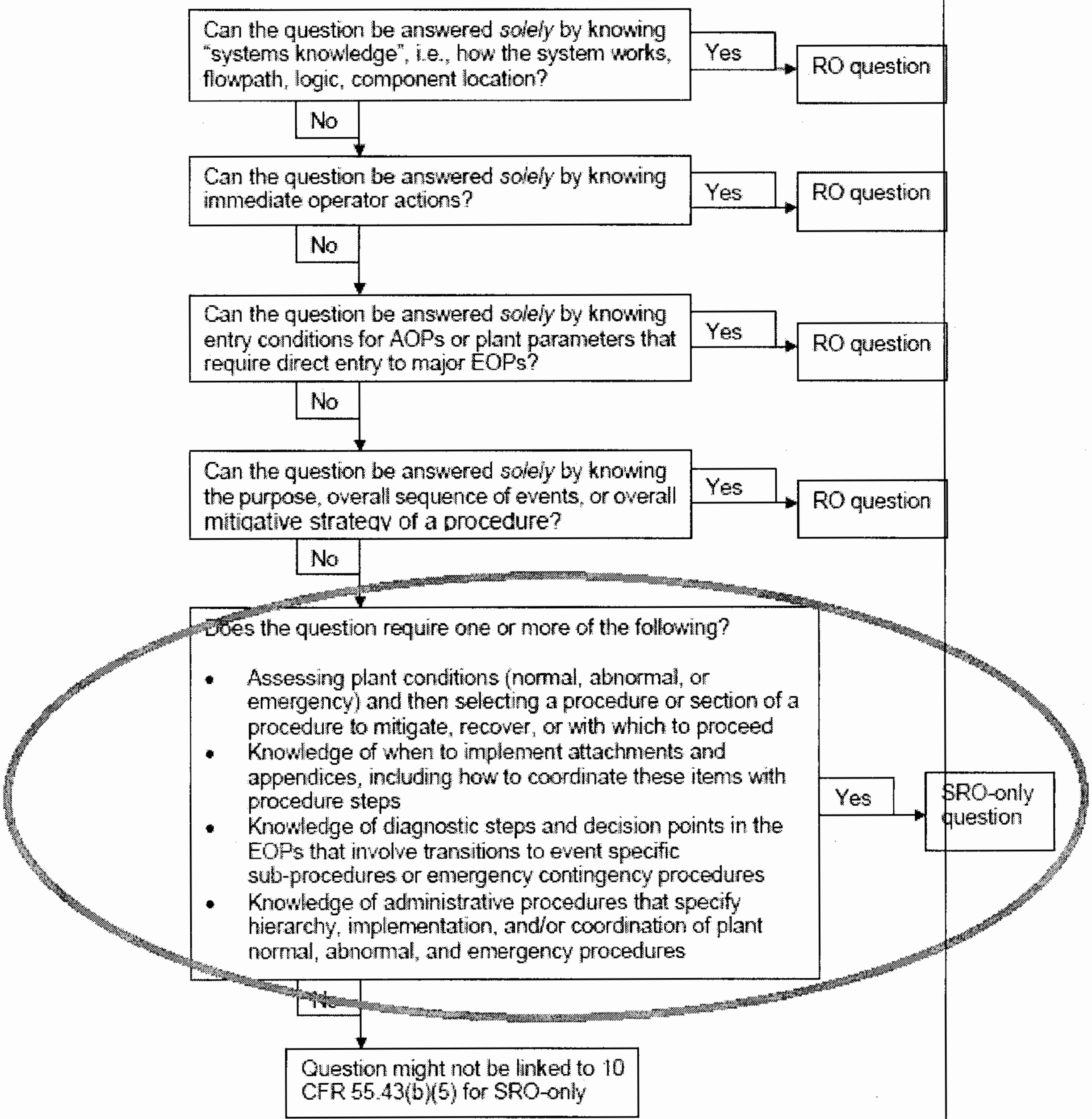
High Cog: This question is high cog because the examinee must know that an RB evacuation is required from the stem.

SRO Only: This question is SRO only because the examinee must assess plant conditions know the content of a procedure to ensure low pressure containment integrity is maintained.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)**




Plant Conditions:

- The plant is shut down in a Refueling Outage.
- Core off-load in progress.
- Containment Integrity is NOT being maintained.

- RB Purge is in progress IAW OP-TM-823-408 RB Purge - RB Doors and/or Equipment Hatch Open.

Event:

- The Fuel Handling SRO reports that a Fuel Assembly has been dropped and damaged in the Refueling cavity.
- MAP C-1-1 RADIATION LEVEL HI , alarms.
- RB Purge Exhaust Duct Monitor RM-A-9 is off-scale high.
- RB Purge Exhaust Duct Monitor RM-A-9 Hi-Hi is 3000 CPM.

Assuming no action other than the initiation of a Containment evacuation has taken place, which ONE (1) of the following identifies the current positions of RB Purge Isolation Valves (AH-V-1A, 1B, 1C and 1D), AND the reason for their position?

A. Open;
The RB Purge Line Isolation High Radiation Interlock is defeated.

B. Closed;
The RB Purge Line Isolation High Radiation Interlock was operated by RM-A-9G when it went above the ALERT level.

C. Closed;
The RB Purge Line Isolation High Radiation Interlock was operated by RM-A-9 Hi-Hi when it went above the ALERT level.

D. Open;
The RB Purge Line Isolation High Radiation Interlock has not operated because RM-A-9 Hi-Hi has not exceeded the HIGH Alarm level.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**91** **ID: 1700311** **Points: 1.00**

Plant conditions:

- The crew is performing a plant startup per 1102-02, Plant Startup.
- Deboration to the Target Critical Boron Concentration has just been completed.
- The crew is awaiting Plant Manager authorization to startup the reactor.

EVENT:

A check of the Backup Incore Thermocouple Readout (BIRO) reveals that the following Thermocouples have failed:

4-E
5-G
7-B
13-C

Given these current conditions, which ONE of the following identifies whether the LCO for Tech Spec 3.5.5, Accident Monitoring Instrumentation, is currently satisfied, and if not, also identifies the MINIMUM actions required to satisfy the LCO without reliance on any action statement?

A. LCO 3.5.5 is met;
No action statements are required to be entered.

B. LCO 3.5.5 is NOT met; AND
Performing maintenance and declaring Thermocouple 13-C OPERABLE will allow all actions statements to be exited.

C. LCO 3.5.5 is NOT met; AND
Performing maintenance and declaring Thermocouple 4-E OPERABLE will allow all actions statements to be exited.

D. LCO 3.5.5 is NOT met; AND
Performing maintenance and declaring Thermocouple 4-E and 7-B OPERABLE will allow all actions statements to be exited.

Answer: C

**Answer Explanation**

| | | |
|---|---|---|
| ****Needs 1105-5 Table 2 as a reference | | |
| Explanation: To answer this question correctly, the examinee must know: (1) That 2 of the 4 instruments per quadrant must be operable or the action statement of Tech Spec 3.5.5 must be entered (2) The condition of the plant is one where the Tech Spec 3.5.5 is applicable (3) How to read Table 2 of 1105-5 to verify that 3 in a quadrant are failed. | | |
| A. | LCO 3.5.5 is met;<br>No action statements are required to be entered. | This is plausible because many LCOs are written to be applicable when the reactor is critical. Since it is not, the operator may incorrectly believe that thermocouples do NOT need to be OPERABLE to satisfy the LCO. |
| B. | LCO 3.5.5 is NOT met; AND Performing maintenance and declaring Thermocouple 13-C OPERABLE will allow all actions statements to be exited. | This is plausible because the operator may incorrectly believe that thermocouple 13-C is part of the same Quadrant as two of the others. The provided reference allows the student to map the BIROs and determine that 3 out of four BIROs in one quadrant are not operable therefore the TS LCO is not met and that 13-C is not one of those three. Therefore, performing maintenance on this thermocouple would still leave the operator with three inoperable thermocouples in a quadrant. |
| C. | LCO 3.5.5 is NOT met; AND Performing maintenance and declaring Thermocouple 4-E OPERABLE will allow all actions statements to be exited. | According to Technical Specification LCO 3.5.5 the instruments identified in Table 3.5-2 and Table 3.5-3 during STARTUP, POWER OPERATION and HOT STANDBY shall be OPERABLE. According to Technical Specification LCO 3.5.5, there are four Backup Incore Thermocouple Display (BIRO) channels/ per quadrant, and two of them must be OPERABLE. According to Technical Specification 1.0 , the reactor shall be considered in the STARTUP mode when the shutdown margin is reduced with the intent of going critical. Since the plant has been heated up to beyond 250°F under the stated conditions, and the Boron concentration has been reduced to Target Critical Boron Concentration, the plant is in the STARTUP mode, and the LCO is applicable. This is supported by the System Operating Procedure. According to 1105-5 when RCS avg. temp is >250°F, the Backup Incore Thermocouple Display Channel (BIRO) must be operable with a minimum of 2 thermocouples per core quadrant in operation. Using the provided reference and generating a core map, the operator can determine that 4-E, 5-G, 7-B and 13-C are in two different quadrants and three (4-E, 5-G and 7-B) are in the same quadrant. Since a minimum of two are required to be OPERABLE in each quadrant, the operator can determine that one quadrant has only one OPERABLE thermocouple; 8-F. This means that LCO 3.5.5 is NOT being met. The operator can meet LCO 3.5.5 by performing maintenance and declaring Thermocouple 4-E OPERABLE. |
| D. | LCO 3.5.5 is NOT met; AND Performing maintenance and declaring Thermocouple 4-E and 7-B OPERABLE will allow all actions statements to be exited. | This is plausible because the operator may incorrectly believe that only one thermocouple per quadrant is allowed to be inoperable. While performing maintenance on both thermocouples will render the LCO met, it can be met with less maintenance, by performing maintenance only on thermocouple 4-E; and therefore, it is NOT the minimum action necessary to meet the LCO. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Group # | | | 2 |
| K/A # | 017 | 2.4.3 | |
| Importance Rating | 3.7 | | 3.9 |

K/A: In-Core Temperature Monitor: Ability to identify post-accident instrumentation

Proposed Question: Question #91

Technical Reference(s): 1105-5, Rev 25

TS 3.5.5, Table 3.5-2

Proposed References to be provided to applicants during examination: 1105-5, Table 2

Learning Objective: 625-GLO-14

Question Source: Bank # 860160

Modified Bank #

New

Question History: Comp 2 Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 b.2

Comments:

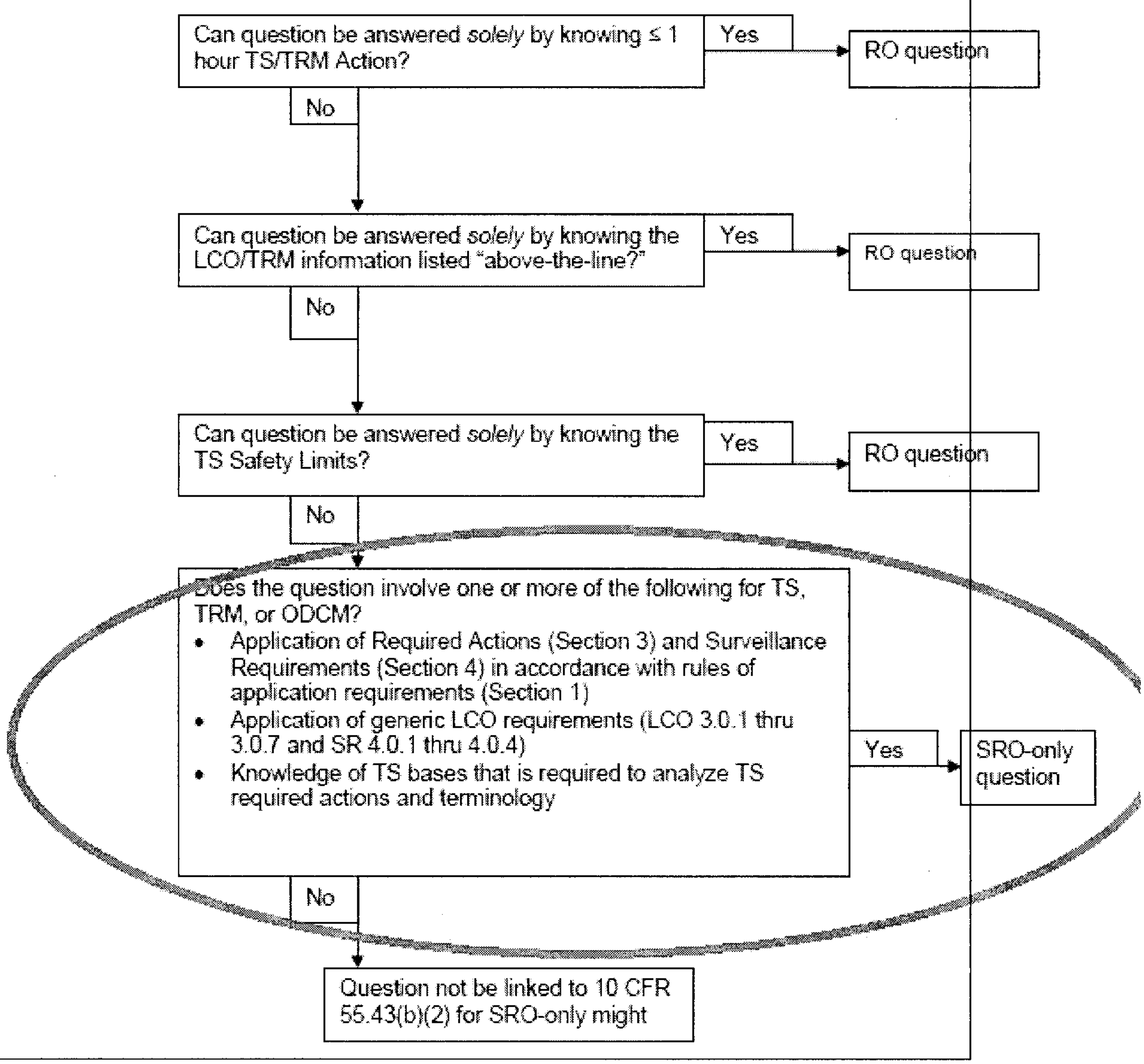
K/A Match: This question matches the K/A because the examinee must know how many thermocouples are required to be operable to meet the TS 3.5.5, which is for post accident monitoring.

The question is at the Comprehension/Analysis cognitive level because the operator must consider several pieces of information such as (1) the LCO governing the BIRO, (2) the condition of the plant and whether or not the LCO is applicable under the present conditions, and (3) the specific instrument failures; and then draw a conclusion regarding the status of the LCO (and the maintenance needed to satisfy it).

The question is SRO-Only because the operator must demonstrate the ability to comply with the plant Technical Specifications including knowing procedurally required minimum requirements for instrument operability .

**ES-401** **5** **Attachment 2**

**Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**92** **ID: 1736504** **Points: 1.00**

Plant conditions:

- 100% power with ICS in full Auto.
- Condensate Pump CO-P-1C is tagged out for maintenance.

EVENT:

- The Secondary NLO reports that vibrations on Condensate Pump CO-P-1A have increased since his last round and a high pitched noise seems to be coming from the lower motor bearing.
- Maintenance confirms that the lower bearing is failing and recommends CO-P-1A be stopped within the next hour to avoid shaft damage.

The operating crew must reduce power to __(1)__, and secure CO-P-1A in accordance with __(2)__.

A. (1) < 665 MWe
(2) 1102-4, POWER OPERATIONS

B. (1) < 665 MWe
(2) OP-TM-421-430, REMOVING CO-P-1A FROM SERVICE

C. (1) < 50% power
(2) 1102-4, POWER OPERATIONS

D. (1) < 50% power
(2) OP-TM-421-430, REMOVING CO-P-1A FROM SERVICE

Answer: C

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) Due to CO-P-1C being out of service, the operating crew must down power the plant in accordance with 1102-4; (2) Also, due to CO-P-1C being out of service, the crew must secure a Main Feedwater Pump and a Condensate Booster Pump prior to securing the Condensate Pump; (3) In accordance with 1102-4, the operating crew could secure the Main Feedpump, Condensate Booster Pump, and the Condensate Pump at less than < 560 MWe, and prior to a Condensate Booster Pump (CO-P-2) discharge pressure exceeding 675 psig; (4) 560 MWe is approximately 68% power; (5) A precaution in OP-TM-421-000 limits operations with only one Condensate Pump (with only one Condensate Booster Pump, and Main Feedwater Pump) to less than 50% power to avoid a Reactor Trip.

| | | |
|---|---|---|
| A. | (1) < 665 MWe<br>(2) 1102-4, POWER OPERATIONS | INCORRECT: Plausible if the examinee believes that the runback setpoint for loss of a Feedwater Pump is 665 MWe. This is incorrect because the plant would run back to less than 560 MWe, in addition this power level would be to high for sustained operation with a single condensate pump in operation. |
| B. | (1) < 665 MWe<br>(2) OP-TM-421-430, REMOVING CO-P-1A FROM SERVICE | INCORRECT: Plausible if the examinee believes that the runback setpoint for loss of a Feedwater Pump is 665 MWe. This is incorrect because the plant would run back to less than 560 MWe, in addition this power level would be to high for sustained operation with a single condensate pump in operation. |
| C. | (1) < 50% power<br>(2) 1102-4, POWER OPERATIONS | Correct Answer: See above. |
| D. | (1) < 50% power<br>(2) OP-TM-421-430, REMOVING CO-P-1A FROM SERVICE | INCORRECT: Plausible because the examinee could believe OP-TM-421-430 contains the correct guidance on securing CO-P-1A. Incorrect because this procedure would start the standby Condensate Pump first, which is out of service in the stem. The guidance to secure the Main Feedwater Pump, Condensate Booster Pump, and Condensate Pump are in 1102-4. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 2 |
| | K/A # | 056 | A2.04 |
| | Importance Rating | | 2.6 |

K/A: Condensate: 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, of mitigate the consequences of those malfunctions or operations: Loss of condensate pumps.

Proposed Question: Question #92

Technical Reference(s): OP-TM-421-000, Rev 16

1102-4, Rev 133

Proposed References to be provided to applicants during examination: None

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

Learning Objective: 421-GLO-10

Question Source: Bank # 572934

Modified Bank #

New

Question History: System Exam 5 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

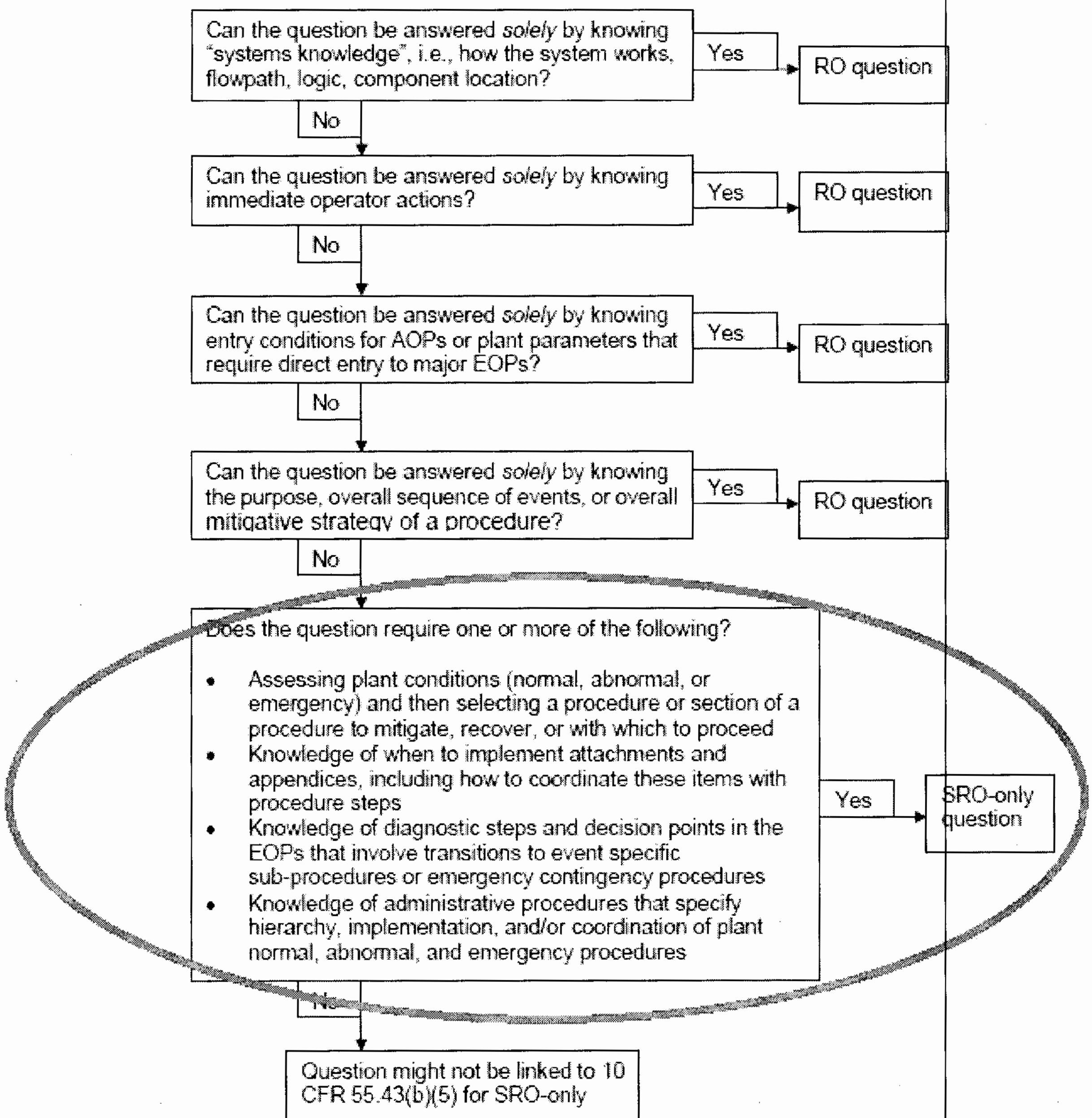
55.43 b.5

Comments:
KA Match: This question matches the KA because the examinee has to understand the impact of losing a second condensate would be to the plant. The examinee must also determine the correct power and procedure that the operators would use to lower power.

High cog: This question is High Cog because the examinee must know that the plant power must be lowered if another condensate pump is lost. In addition, the examinee must pick the correct procedure to lower power.

SRO Only: This question requires the examinee to assess the plant conditions/equipment and to know the content of procedures in order to select the required course of action.

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5) (Assessment and selection of procedures)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**93** **ID: 1736533** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.
- An approved radioactive liquid release is in progress.

EVENT:

- Annunciator C-1-1 “RADIATION LEVEL HI” is in alarm.
- Radiation Monitor RM-L-6, RAD WASTE DISCHARGE, has failed (pegged high, off-scale).
- WDL-P-14A/B discharge valve to the MDCT, WDL-V-257, is stuck open and cannot be closed.

Identify the one selection below that describes:
(1) The correct action, and
(2) The timeclock associated with RM-L-6.

A. (1) Terminate the release and notify Radiation Protection.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 14 days.

B. (1) Terminate the release and notify Radiation Protection.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 30 days.

C. (1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 14 days.

D. (1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations.
(2) Exert best efforts to return RM-L-6 to OPERABLE within 30 days.

Answer: B

**Answer Explanation**

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<Explanation: To answer this question correctly, the examinee must know: (1) In accordance with OP-TM-MAP-C0101, when RM-L-6 alarm is received (in this case for RM-L-6 failing high), WDL-V-257 should close; (2) When the valve (WDL-V-257) does not close, and because RM-L-6 is failed high, the examinee should determine that RM-L-6 is inoperable; (3) OP-TM-MAP-C0101 directs the termination of the release and the notification of radiation protection; (4) The timeclock to return RM-L-6 is found in CY-TM-170-300, OFFSITE DOSE CALCULATION MANUAL (ODCM) step 2.1.1, where with less than the number OPERABLE best efforts to return the instrumentation to OPERABLE within 30 days, if not explain in the next Annual Effluent Release Report.

| | | |
|---|---|---|
| A. | (1) Terminate the release and notify Radiation Protection.<br>(2) Exert best efforts to return RM-L-6 to OPERABLE within 14 days. | INCORRECT - Plausible because many ODCM timeclocks are 14 days. Incorrect because the timeclock from RM-L-6 is 30 days. |
| B. | (1) Terminate the release and notify Radiation Protection.<br>(2) Exert best efforts to return RM-L-6 to OPERABLE within 30 days. | Correct Answer - See above. |
| C. | (1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations.<br>(2) Exert best efforts to return RM-L-6 to OPERABLE within 14 days. | INCORRECT - Plausible because this is the action from OP-TM-MAP-C0101 if RM-L-7 (Plant Discharge Rad Monitor) high alarm is in. In addition, the timeclock is 30 days, not 14 days. |
| D. | (1) Request Chemistry obtain a sample of the Rad Monitor Pit and check the permit calculations.<br>(2) Exert best efforts to return RM-L-6 to OPERABLE within 30 days. | INCORRECT T - Plausible because this is the action from OP-TM-MAP-C0101 if RM-L-7 (Plant Discharge Rad Monitor) high alarm is in. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| | Group # | | 2 |
| | K/A # | 068 | A2.04 |
| | Importance Rating | | 3.3 |

K/A: Liquid Radwaste: Ability to (a) predict the impacts of the following malfunctions or operations on the Liquid Radwaste System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of automatic isolation.

Proposed Question: Question #93

Technical Reference(s): OP-TM-MAP-C0101, Rev 3

CY-TM-170-300, Rev 4

Proposed References to be provided to applicants during examination: None

Learning Objective: 232-GLO-14

Question Source: Bank # 1102535

Modified Bank #

New

Question History: System Exam 14 Last NRC Exam: 10-02 NRC

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 b.2

Comments:

KA Match: This question matches the KA because the examinee must know that if RM-L-6 fails high, that the release must be terminated and RP notified. To mitigate the failure, the examinee must know that the site has 30 days to repair the detector or the reason for the extended inoperability must be included in the next Annual Effluent Release Report.

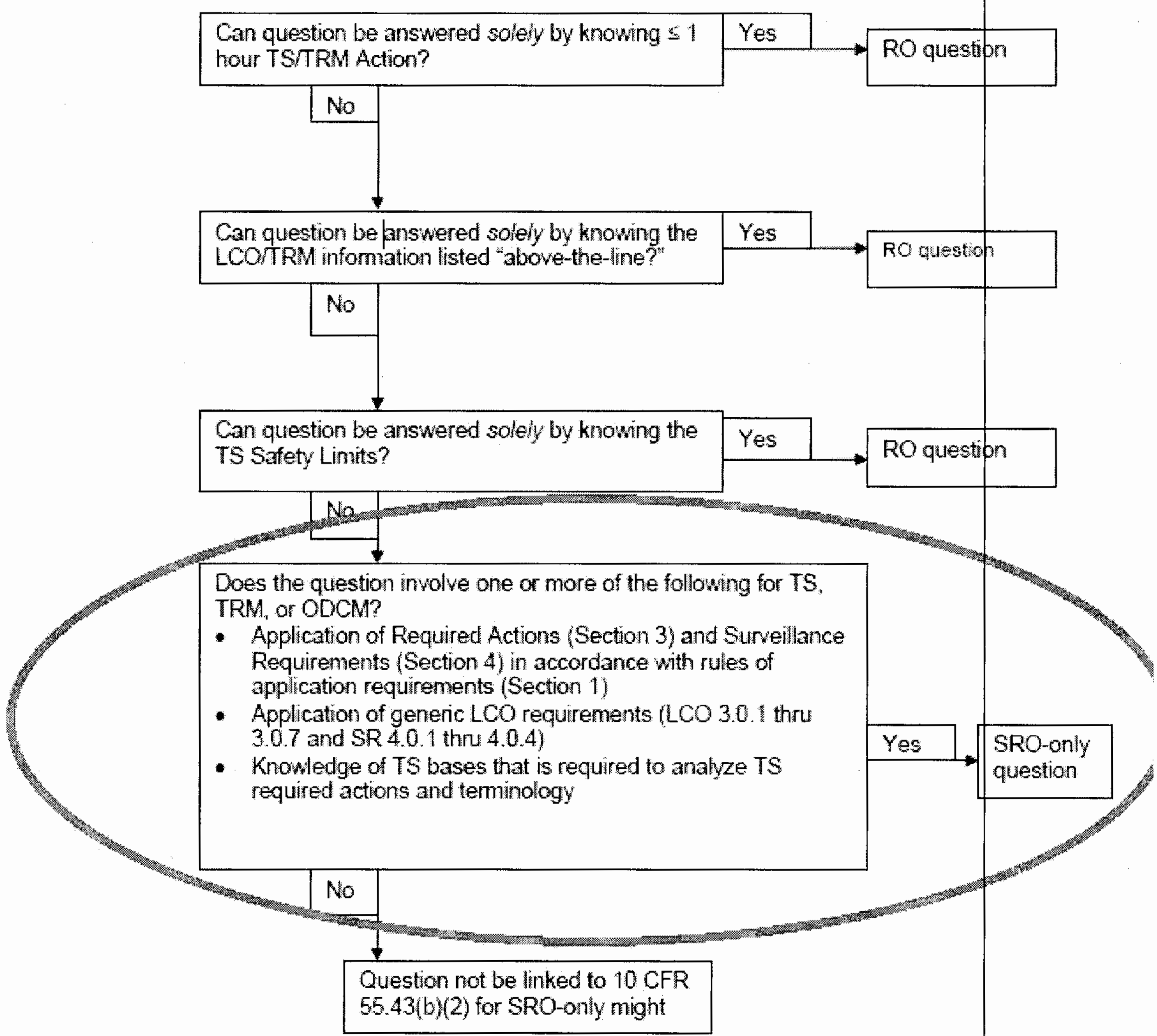
SRO only: This question is SRO only because the examinee must know how to apply the action of the ODCM for RM-L-6 inoperability.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM



**Figure 1: Screening for SRO-only linked to 10 CFR 55.43(b)(2) (Tech Specs)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**94** **ID: 1700144** **Points: 1.00**

Plant Conditions:

- 100% power with ICS in full auto.

Per Technical Specifications, which of the following is true regarding makeup tank level and pressure?

Extended operation inside the __(1)__ is not permitted due to prevention of __(2)__.

A. (1) Restricted Region
(2) a loss of NPSH when 2 Makeup Pumps are running.

B. (1) Low NPSH region
(2) a loss of NPSH when 2 Makeup Pumps are running.

C. (1) Restricted Region
(2) gas entrainment into the makeup pumps in the event of an emergency injection on a Large Break LOCA.

D. (1) Low NPSH region
(2) gas entrainment into the makeup pumps in the event of an emergency injection on a Large Break LOCA.

Answer: C

**Answer Explanation**

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

<<<<Explanation: To answer this question correctly, the examinee must know: (1) Operation in the LOW NPSH Region of Attachment 7.3 of OP-TM-211-000 is prohibited to ensure that adequate MU pump NPSH during the limiting design basis event; (2) Operation is not permitted in the "Prohibited" or "Restricted" Region to prevent gas entrainment into the makeup pumps in the event of emergency injection on a Large Break LOCA. The ensures that the MU tank gas bubble remains in the MU tank as long as the BWST level is above minimum.

| | | |
|---|---|---|
| A. | (1) Restricted Region<br>(2) a loss of NPSH when 2 makeup pumps are running. | Plausible because the restricted region is in a area where low makeup tank level and pressure are achievable (i.e 50" in the MU tank and 15 psig) hence the examinee could believe the restricted region area of concern is NPSH. Incorrect because with the BWST lowering, operation in the restricted and prohibited region is the primary concern. Due to a lower BWST level, the differential pressure between it and the makeup tank and is lower. The casualty which makes this become an issue is a Large Break LOCA, which requires inventory supplied by the makeup pumps. At a lower differential pressure, and with potentially all three makeup pumps running, the suction source could be either the makeup tank or the BWST (depending on which has the higher pressure). If the conditions existed where the makeup tank has the higher pressure than the BWST in conjunction with a low MU tank level, gas could enter the makeup pump suction line making all three makeup pumps inoperable. |
| B. | (1) Low NPSH region<br>(2) a loss of NPSH when 2 makeup pumps are running. | Plausible because this is the correct basis for prevention of operation in the low NPSH region. Incorrect because with a lowering BWST level, operation in the prohibited region is of concern. |
| C. | (1) Restricted Region<br>(2) gas entrainment into the makeup pumps in the event of an emergency injection on a Large Break LOCA. | Correct Answer. |
| D. | (1) Low NPSH region<br>(2) gas entrainment into the makeup pumps in the event of an emergency injection on a Large Break LOCA. | Plausible because the examinee could believe that with a Low NPSH, that gas entrainment is of concern during and emergency injection on a LOCA is a concern. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 1 |
| | K/A # | 2.1.32 | |
| | Importance Rating | | 4.0 |

K/A: Ability to explain and apply system limitations and precautions.

Proposed Question: Question #94

Technical Reference(s): OP-TM-211-000, Rev 33

T.S. 3.3 basis

Proposed References to be provided to applicants during examination: None

Learning Objective: 211-GLO-6

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 b.2

Comments:

K/A match: This is a K/A match because it is a precaution and limitation of the makeup system (OP-TM-211-000).

SRO Only: This is SRO only because the examinee must identify and recall the basis behind the MU tank pressure and level curves described in T.S. 3.3.1.1 g.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**95** **ID: 1700178** **Points: 1.00**

Plant conditions:

- The plant is in refueling outage with fuel handling operations in the RB in progress
- The Main Bridge Operator has lowered an assembly that is NOT an anti-straddle type assembly into the core location with open water on one side only.
- The ZZ tape reading for the assembly is approximately 2" high with full down load cell indication

With the above indications the assembly is resting on the ________________.

A. grid of an adjacent assembly and can be shaken while moving through the grid region as the assembly is lowered

B. grid of an adjacent assembly and the mast can be rotated to reposition the assembly prior to lowering

C. reactor lower grid and can be moved slightly toward the open side and lowered

D. reactor lower grid and the assembly can be raised and jog can be used to reposition the assembly prior to lowering

Answer: D

**Answer Explanation**

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

Explanation: To answer this question correctly, the examinee must know: (1) the likely cause (and the only possible cause as determined by the stem and distractors) is that the assembly is resting on the reactor lower grid; (2) since the assembly is not boxed in and is NOT the anti-straddle design, the assembly should NOT be moved toward the open location; (3) The four allowable actions are to: Raise the F/A until full weight and try to re-lower, use jog to reposition the F/A, rotate mast or flow mast by pulling deten pin as required, shake the fuel hoist cable.

| | | |
|---|---|---|
| A. | grid of an adjacent assembly and can be shaken while moving through the grid region as the assembly is lowered | Plausible since shaking a fuel assembly while moving is allowed; however not through the grid region. |
| B. | grid of an adjacent assembly and the mast can be rotated to reposition the assembly prior to lowering | Plausible since rotating the mast is allowed; however the assembly is not resting on an adjacent assembly grid it is resting on the reactor lower grid. |
| C. | reactor lower grid and can be moved slightly toward the open side and lowered | Plausible since the assembly is resting on the reactor lower grid; however moving the assembly toward the open area is not allowed. |
| D. | reactor lower grid and the assembly can be raised and jog can be used to reposition the assembly prior to lowering | Correct answer. The assembly is on the reactor lower grid and can be raised and jog can be used to realign the assembly IAW section 7.4 of 1505-3. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 1 |
| | K/A # | 2.1.42 | |
| | Importance Rating | | 3.4 |

K/A: Knowledge of new and spent fuel movement procedures.

Proposed Question: Question #95

Technical Reference(s): 1505-3, Rev 23

Proposed References to be provided to applicants during examination: None

Learning Objective: N/A

Question Source: Bank # 718417

Modified Bank #

New

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

Question History: Comp 3 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 b.7

Comments:

K/A Match: The matches the K/A because the examinee must have knowledge of the fuel handling procedures and the SRO responsibilities in those procedures.

SRO Only: This question is SRO only because it is the responsibility of the Fuel Handling Supervisor to attempt to direct the reseat the fuel assembly by one of the approved methods in the procedure (1505-3).

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**96** **ID: 1700737** **Points: 1.00**

Plant Conditions:

- Your shift has just assumed the watch in the Control Room.
- The Special Test Coordinator (a designee of the Senior Line Manager) has just commenced conducting a brief for Turbine Torsional Testing, which is scheduled to take place this shift.
- Due to one of the maintenance team members calling out sick, the testing group is short-handed.
- The Special Test Coordinator is briefing the following actions:
  - The Special Test Coordinator will assume the active role of the performer who called out sick.
  - Just in Time training is not required because the evolution was performed by the same team last time, so the members understand the actions required.
  - Station Management shall be present during the entire test.
  - The test is scheduled to take more than one shift and another brief will be required upon turnover.

In accordance with OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTIONS, which one of the following is a true statement with regards to the actions that were briefed?

A. The Special Test Coordinator can not be an active performer of the test.

B. Station Management does not need to be present during the special test.

C. Only a Senior Line Manager can conduct the briefing for the special test.

D. Just in Time training must occur to ensure the test is understood by those involved.

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) A caution in section 4.3 of OP-AA-108-110 states the Test Coordinator is not an active performer of the test. They provide oversight for the test activities. They are coordinators and the interface with Operations during the test.

| | | |
|---|---|---|
| A. | The Special Test Coordinator can not be an active performer of the test. | Correct Answer |
| B. | Station Management does not need to be present during the special test. | Incorrect - Step 4.3.5 requires Senior Line Management and the Test Coordinator to be present prior to and during the special test or evolution performance. This distractor is plausible because with most complex evolutions, the Station Management only has to be present for the prejob brief. |
| C. | Only a Senior Line Manager can conduct the briefing for the special test. | Incorrect - Step 4.3.3 - The SLM shall ENSURE an IPA briefing is conducted in accordance with HU-AA-1211 prior to performing the special test or evolution. This does NOT mean the that SLM must conduct the brief. This is plausible because the examinee could believe the SLM must conduct the brief. |
| D. | Just in Time training must occur to ensure the test is understood by those involved. | Incorrect - Step 4.3 - Consider Just in Time training to ENSURE plant conditions and the activity are understood by those involved. In addition OP-AA-101-111-1001 states JIT must be considered only. JIT training is not required to be performed. Plausible because the examinee could believe it is required by this procedure of OP-AA-101-111-1001. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | | 3 |
| Group # | | 2 |
| K/A # | 2.2.7 | |
| Importance Rating | | 3.6 |

K/A: Knowledge of the process for conducting special or infrequent tests.

Proposed Question: Question #96

Technical Reference(s): OP-AA-108-110, Rev 3

Proposed References to be provided to applicants during examination: None

Learning Objective: AOP-DBIG-PCO-3

Question Source: Bank # 742715

Modified Bank #

New

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| Question History: | Sim Exam 5 | Last NRC Exam: | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | X |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | b.5 | |

Comments:

KA Match: This matches the KA because the examinee will have to have knowledge of OP-AA-108-110, EVALUATION OF SPECIAL TESTS OR EVOLUTION procedure.

SRO Only: This is an SRO only question because the question tests the examinees knowledge of a procedure that has the purpose of providing a mechanism for evaluation of infrequently performed, complex tasks to implement special administrative management controls. The examinee will have to understand the hierarchy and roles of personnel during the infrequently performed, complex task.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**97** **ID: 1700778** **Points: 1.00**

Sequence of Events:

- An In-service Test (IST) performed on Nuclear River (NR) system pump NR-P-1A showed a flow rate of 6234 gpm, which is less than the minimum flow rate allowed by Technical Specification.
- Subsequent testing performed under a complex troubleshooting plan had shown:
  - Pump performance appeared to have leveled out at a lower flow rate, but still greater than the design minimum ASME value.
  - Engineering evaluations concluded that there would be no further degradation of flow rate.

After the troubleshooting plan is complete, NR-P-1A must be declared __(2)__.

A. Operable

B. Degraded

C. Inoperable

D. Unavailable

Answer: C

| **Answer Explanation** |
|---|

Explanation: To answer this question correctly, the examinee must know: (1) From OP-AA-108-104, TECHNICAL SPECIFICATION COMPLIANCE, the definition of INOPERABLE is when an SSC is considered to be INOPERABLE when it is not capable of meeting all of the requirements of the Technical Specification, ATR, TRM, ISFSI, or ODCM definition for OPERABILITY; (2) Even though Engineering has determined that the pump will not degrade further, and that the flow is greater than the ASME value, this does not make the pump anything other than INOPERABLE because it is still lower than the minimum flow rate allowed by technical specifications.

| | | |
|---|---|---|
| A. | Operable | Incorrect but plausible because the definition of OPERABLE is the SSC is capable of performing its specified function. In addition, there is clarification that the deficiency which caused the equipment to be inoperable has been resolved. The stem states that evaluations have been completed that there will be no further degradation, this may imply, but does not explicitly say that the deficiency is resolved. |
| B. | Degraded | Incorrect, but plausible because the term degraded only applies when the equipment is OPERABLE. The examinee may believe that the equipment is not INOPERABLE because the SSC can still meet ASME value, and will not degrade further, but since is below that technical specification value that the SSC is degraded. |
| C. | Inoperable | Correct Answer |
| D. | Unavailable | Plausible because if this were a 2 pump system, this malfunction would be tracked under for unavailability. Incorrect because when NR-P-1A failed the T.S surveillance, NR-P-1B would have been ES selected on the 1T 480V bus. |

Examination Outline Cross-reference:

| Level | RO | SRO |
|---|---|---|
| Tier # | | 3 |
| Group # | | 2 |
| K/A # | 2.2.21 | |
| Importance Rating | | 4.1 |

K/A: Knowledge of pre- and post-maintenance operability requirements

Proposed Question: SRO Question #22

Technical Reference(s): OP-AA-108-104
ER-TM-310-1001

Proposed References to be provided to applicants during examination: None

Learning Objective: 108104-APCO-1

Question Source: Bank # 746855
Modified Bank #

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

| | | | |
|---|---|---|---|
| | New | | |
| Question History: | Sim Exam 3 | Last NRC Exam: | N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | | |
| | 55.43 | b.2 | |

Comments: (KA Match, why high cog, why SRO only)

K/A Match: This question matches the KA because the examinee must make the determination that NR-P-1A is inoperable even though the flow rate is greater than the ASME code and engineer has determined no further degradation will occur.

SRO Only: This question is SRO only because the examinee must determine the operability of an ES pump based on the data given.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**98** **ID: 1700579** **Points: 1.00**

Both evaporators are out of service

Operations has initiated a liquid waste release permit in accordance with CY-TM-170-2001, RELEASING RADIOACTIVE LIQUID WASTE for the a WESCT.

The initial sample radioactivity was over 7E-7 uCi/ml (excluding tritium and noble gases).

Who one of the following is correct regarding the release of the WESCT?

A. The tank must be reprocessed and then released with Shift Manager approval.

B. The tank must be reprocessed and then released with Shift Manager and Chemistry Management approval.

C. The tank can be released with Shift Managment approval, ONLY.

D. The tank can be released with BOTH Shift Manager and Chemistry Management approval.

Answer: D

**Answer Explanation**

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

WECST = Waste Evaporator Condensate Storage Tank (WDL-T-11A/11B). Procedures refer to the tanks as WECSTs, therefore WECST is not defined in the question to avoid confusion.

Explanation: To answer this question correctly, the examinee must know: (1) who approves the paperwork generated for a release of a WESCT to the MDCT then to the Susquehanna River; (2) If the total chemistry is above 7E-7 uCi/ml, then Chemistry Management approval is required, but since it is not, so the Shift Manager is the only approval required.

| | | |
|---|---|---|
| A. | The tank must be reprocessed and then released with Shift Manager approval. | Incorrect: The tank contents could be released without re-processing. In addition, since both evaporators are out of service, this may not be possible. |
| B. | The tank must be reprocessed and then releaased with Shift Manager and Chemistry Management approval. | Incorrect: The tank contents could be released without re-processing. In addition, since both evaporators are out of service, this may not be possible. |
| C. | The tank can be released with Shift Managment approval, ONLY. | Incorrect: The Shift Manager must have Chemistry Management approval if above the limit of 7E-7 uCi/ml. |
| D. | The tank can be released with BOTH Shift Manager and Chemistry Management approval. | Correct Answer: See above. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 3 |
| | K/A # | 2.3.6 | |
| | Importance Rating | | 3.8 |

K/A: Ability to approve release permits.

Proposed Question: Question #98

Technical Reference(s): CY-TM-170-2001, Rev 0

Proposed References to be provided to applicants during examination: None

Learning Objective: 232-GLO-9

Question Source: Bank #

Modified Bank # 770815

New

Question History: Unmodified Sim Exam 4 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 b.4

Comments: (KA Match, why high cog, why SRO only)

K/A match: This matches the KA because the examinee must know who approves the radioactive release paperwork.

SRO Only: This is an SRO only question because the examinee must know the process for gaseous/liquid release approvals. The operations portion of a release is to initiate paperwork, approve the paperwork and then perform the release.

IAW CY-TM-170-2001 Releasing Radioactive Liquid Waste, who is responsible for the **Final** Approval on Liquid Permit Pre-Release Report for the release of a WECST?

A. Shift Manager.

B. Chemistry Tech

C. Operations Director

D. Chemistry Management

Answer: A

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**99** **ID: 1685047** **Points: 1.00**

Plant Conditions:

- Reactor is at 100% power.
- A CODE WHITE has been declared by site security

EVENT:

- All communications between the ARO and control room have been lost
- CODE BLUE is declared and confirmed by site security
- CRS/URO continue with OP-TM-AOP-008, SECURITY THREAT / INTRUSION actions in the control room.

(1) When will the ARO start taking action after reporting to the 1E 4160V bus?
(2) What actions will the ARO take?

A. (1) When EF-P-2B is running
(2) Trip MU-P-1B, start MU-P-1A, and ensure seal injection is not lost to the reactor coolant pumps

B. (1) When EF-P-2B is running
(2) Take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to both OTSG's

C. (1) When the control room is breached
(2) Trip MU-P-1B, start MU-P-1A, and ensure seal injection is not lost to the reactor coolant pumps

D. (1) When the control room is breached
(2) Take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to both OTSG's

Answer: B

**Answer Explanation**

Explanation: For the examinee to answer this question correctly, they will have to realize (1) on a CODE WHITE, the ARO is dispatched to the 1E 4160V bus to wait and monitor the EF-P-2B breaker (IAW Attachment 5) (2) when the CODE BLUE is declared, the control room will trip the reactor, trip all RCP's, trip the CW pumps, and open the feeder breaker to the 1G bus (3) when the RCP's are tripped, that will initiate EFW, thus closing the EF-P-2B breaker (4) since communications to the control room have been lost, and EF-P-2B breaker is closed, the ARO will continue Attachment 5 (5) the actions in attachment 5 will have the ARO take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to the OTSG's.

| | | |
|---|---|---|
| A. | (1) When EF-P-2B is running<br>(2) Trip MU-P-1B, start MU-P-1A, and ensure seal injection is not lost to the reactor coolant pumps | Plausible because in the actions described in the second part of the question are actions the URO will take in the control room in ATTACHMENT 1. |
| B. | (1) When EF-P-2B is running<br>(2) Take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to both OTSG's | Correct Answer |
| C. | (1) When the control room is breached<br>(2) Trip MU-P-1B, start MU-P-1A, and ensure seal injection is not lost to the reactor coolant pumps | Plausible because one of the conditions for the ARO to take actions is a Control Room breach. While the CODE BLUE or CODE YELLOW could imply such action, since the CR communications have been lost, the ARO at the 1E 4160V bus would use the indication of EF-P-2B breaker being closed as their indication to proceed. |
| D. | (1) When the control room is breached<br>(2) Take control of EG-Y-1B, block the PORV closed, and ensure EFW flow to both OTSG's | Plausible because one of the conditions for the ARO to take actions is a Control Room breach. While the CODE BLUE or CODE YELLOW could imply such action, since the CR communications have been lost, the ARO at the 1E 4160V bus would use the indication of EF-P-2B breaker being closed as their indication to proceed. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 1 |
| | K/A # | 2.4.34 | |
| | Importance Rating | | 4.1 |

K/A: Knowledge of RO tasks performed outside the main control room during and emergency and the resultant operational effects

Proposed Question: Question #99

Technical Reference(s): OP-TM-AOP-008, Rev 16

OP-TM-AOP-0081, Rev 14

Proposed References to be provided to applicants during examination: None

Learning Objective: EOP-DBIG-PCO-2

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

8

Question Source: Bank #

Modified Bank #

New X

Question History: N/A Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 b.5

Comments:

K/A match: This question matches the K/A because OP-TM-AOP-008 coordinates operators outside of the control room. The CRS/URO will have to know what the ARO will be doing and when at the 1E 4160V bus. The operational effect of procedure is to have control of a diesel generator, ensure RCS inventory is sufficient and have flow to both OTSG's.

High Cog: This question is High Cog because it requires various pieces of information to be analyzed to understand when, and what actions the ARO should be taking.

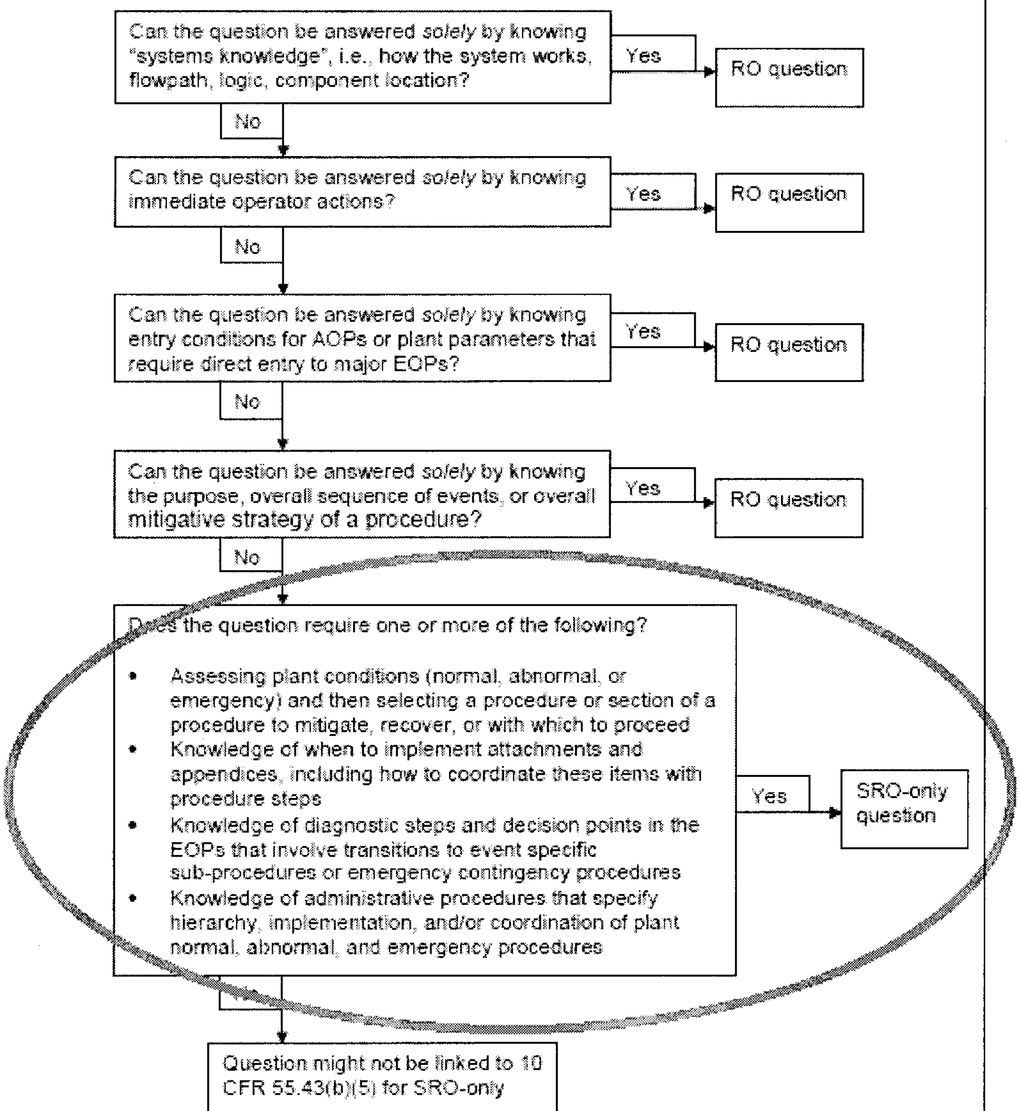
SRO Only: This is an SRO ONLY question because the examinee needs to know when the ARO will be implementing the attachment in OP-TM-AOP-008, and how it is coordinated with other aspects of the Security procedure.

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM



**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)**
**(Assessment and selection of procedures)**



# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

**100** **ID: 1700400** **Points: 1.00**

Plant Conditions:

- Reactor is operating at 100% power, with ICS in full automatic.
- You are the on-shift Control Room Supervisor.
- You have been relieved by the Shift Manager and are returning to the Control Room from the Operations Office Building (OOB).

EVENT:

Just as you pass through the Control Room entrance door:

- One-half of the overhead lights go out, and do NOT come back on.
- You observe the following MAPs actuate:
  - B-1-1, 4KV ES FDR BKR TRIP.
  - B-2-1, 4KV ES BUS UV/OV.
  - B-1-2, 4KV ES MOTOR TRIP.
  - F-1-5, RCP SEAL TOT INJECT FLOW HI/LO.

Based on these conditions, identify the ONE selection below that describes:

(1) The requirements for you to assume a management role in response to the upset, and

(2) The procedure that must be implemented to perform actions that are most critical to mitigation of the event.

A. (1) Obtain a brief update from the shift manager PRIOR to directing team activities.
(2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS.

B. (1) Obtain a brief update from the shift manager PRIOR to directing team activities.
(2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP.

C. (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.
(2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS.

D. (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.
(2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP.

Answer: A

**Answer Explanation**

<<Explanation: To answer this question correctly, the examinee must know: (1) If a member of the control room team is absent when a transient procedure is entered, the team member receives a short briefing from ANY team member before taking ANY directed action; (2) With the indications present, the examinee must interpret that the loss of the 1E 4160V bus has occurred, (e.g., 1/2 control room lights off, Seal injection flow low indicates loss of the running makeup pump on the 1E 4160V bus), and (3) prioritize the loss of the 1E 4160V bus AOP over the listed MAP alarm responses; (4) In accordance with OS-24 section 4.1.5.B, the CRS selects the action significant to overall event mitigation, which would be the loss of the 1E 4160 V bus.

| | | |
|---|---|---|
| A. | (1) Obtain a brief update from the shift manager PRIOR to directing team activities.<br>(2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS. | Correct answer - Even though the on CRS on watch is returning to the control room, they should get an update before directing activities. Even though some of the alarms listed have actions in the alarm responses, the controlling document would be the AOP entered due to the loss of the 1E 4160V bus. |
| B. | (1) Obtain a brief update from the shift manager PRIOR to directing team activities.<br>(2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP. | Plausible because MAP B-1-2 is a cherry alarm on the top row of its section, which indicates it is a priority alarm. The examinees are trained that the general order of priority are the cherry alarms, followed by alarms in the 1st row of every MAP section. In addition, the MAP B-1-2 alarm would NOT direct the examinee to the AOP for a loss of the 1E 4160V bus. |
| C. | (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.<br>(2) OP-TM-AOP-014, LOSS OF 1E 4160V BUS. | Plausible because the returning CRS is a watchstander in command of the control room during abnormal events. Incorrect because OS-24 directs the CRS to receive an update prior to directing activities. |
| D. | (1) Formally announce to the team that you are re-assuming the role of CRS, and THEN begin directing team activities.<br>(2) Alarm Response Procedure for MAP B-1-2, 4KV ES MOTOR TRIP. | Plausible because the returning CRS is a watchstander in command of the control room during abnormal events. Incorrect because OS-24 directs the CRS to receive an update prior to directing activities. In addition, plausible because MAP B-1-2 is a cherry alarm on the top row of its section, which indicates it is a priority alarm. The examinees are trained that the general order of priority are the cherry alarms, followed by alarms in the 1st row of every MAP section. In addition, the MAP B-1-2 alarm would NOT direct the examinee to the AOP for a loss of the 1E 4160V bus. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| | Group # | | 4 |
| | K/A # | 2.4.45 | |
| | Importance Rating | | 4.3 |

K/A: Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: Question #100

# EXAMINATION ANSWER KEY

16-01 SENIOR REACTOR OPERATOR NRC EXAM

Technical Reference(s): OS-24, Rev 28

OP-TM-AOP-014

Proposed References to be provided to applicants during examination: None

Learning Objective: 63501001

Question Source: Bank # 371264

Modified Bank #

New

Question History: Sim Exam 6 Last NRC Exam: N/A

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 b.5

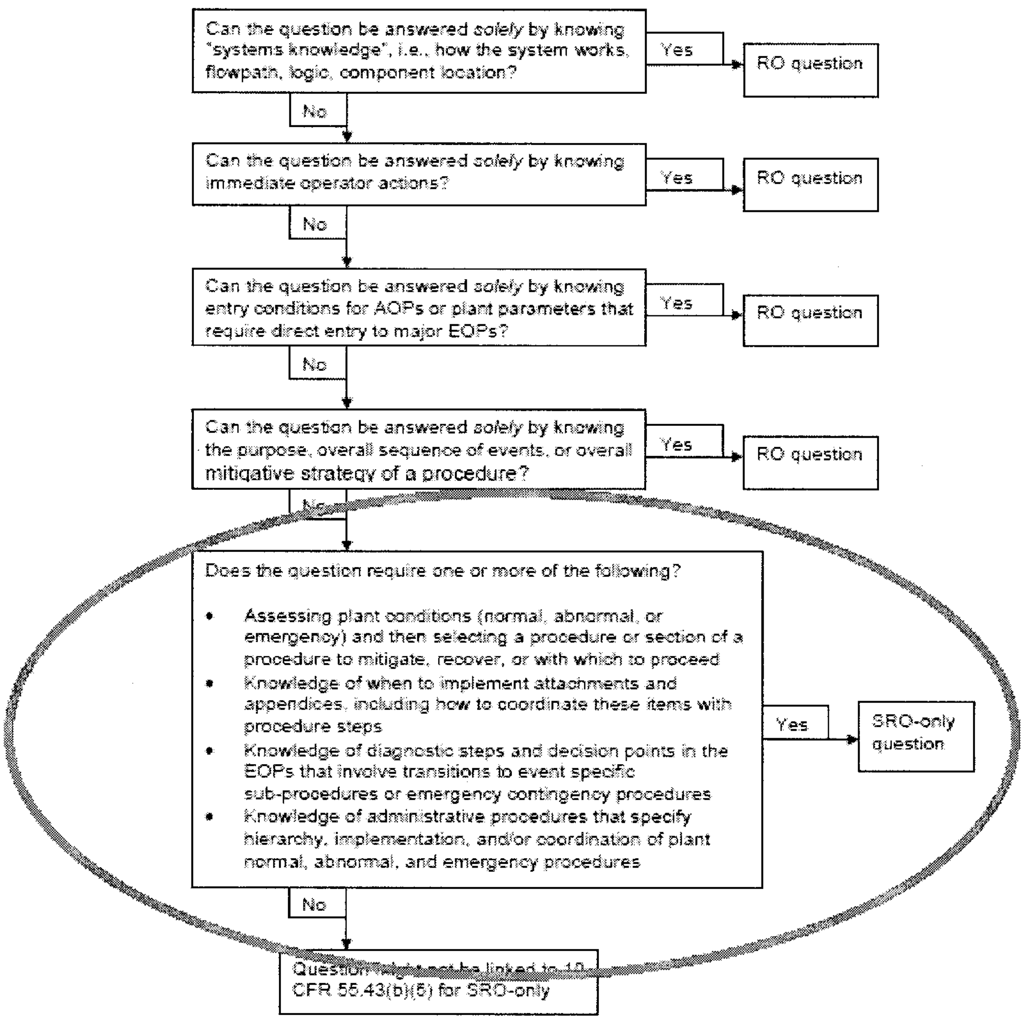
Comments:

K/A Match: This matches the K/A because the examinee has to analyze the given conditions and alarms, and have knowledge of OS-24 to understand that it is the CRS positions job to select the procedure or alarm response to mitigate the event.

High Cog: This is high cog because the examinee must analyze the event and alarms, and choose the correct procedure to mitigate the event.

SRO Only: This is SRO only because the examinee must have knowledge of the administrative procedure OS-24 to understand how to prioritize the alarms and AOP entry.

**Figure 2: Screening for SRO-only linked to 10 CFR 55.43(b)(5)**
**(Assessment and selection of procedures)**

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