EXAM MATERIAL

Hope Creek

Written Exam Analysis

FINAL

9/17/2007

Question: 1 Answer: D

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1 Pt(s) The Unit is operating at 100% power near the end of core life, and the OPRM system is OPERABLE. Turbine Auxiliaries Cooling System (TACS) temperatures are elevated due to high summertime ambient conditions. "A" Recirc MG Set lube oil temperature rises to 206 F, and "B" Recirc MG Set lube oil temperature rises to 213 F.

Which ONE of the following describes the status of the Reactor Recirc Pumps AND the required operator action?

A. **Both Recirc Pumps are tripped. Immediately LOCK the Mode Switch in SHUTDOWN to avoid thermal hydraulic instability.**

B. **"A" Recirc Pump is unaffected, and "B" Recirc Pump is tripped. Reduce speed on the "A" Recirc Pump to 48% in order to prevent Jet Pump vibration in the idle loop.**

C. **Both Recirc Pumps are unaffected. Adjust cooling water to restore oil temperatures within the normal band.**

D. **"A" Recirc Pump is unaffected, and "B" Recirc Pump is tripped. Reduce power by inserting control rods as necessary and monitor for power oscillations.**

Distracter Analysis:

A. **Incorrect**: "A" pump does not trip.
Plausible:

B. **Incorrect**: Pump status is correct, but reducing flow to 48% actually causes jet pump vibration in the idle loop, and reducing flow further may cause thermal hydraulic instability.
Plausible:

C. **Incorrect**: "B" drive motor breaker trips with lube oil temperature greater than 210 F.
Plausible:

D. **Correct**: A drive motor trip occurs at 210 F lube oil temperature. "A" is unaffected since the trip set point has not been exceeded. "B" trips since the temperature is above the set point. HC.OP-AB.RPV-003 Condition A directs operators to insert control rods to clear the APRM Upscale alarms and monitor for power oscillations.
Plausible:

Level: RO Exam

Lesson Plan Objective: RECCONE026
(R) Recall the immediate operator actions required for a recirc pump trip IAW HC.OP-AB.RPV-0003, Recirculation System/ Power Oscillations.

Source: New

Level of knowledge: Analysis

Reference(s):
HC.OP-AB.RPV-003(Q), "Recirculation System / Power Oscillations"

KA: 295001.K1.04
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : (CFR: 41.8 to 41.10) AK1.04
†Limiting cycle oscillation: Plant-Specific 2.5 3.3

Comments / Change Record:

- Deleted the word "best" from the stem. (licensee 07/16)
- Replaced the word "appropriate" operator action with "required" operator action in the stem. (licensee 07/16)
- Licensee requested Power / Flow map as a reference, but the request was **denied** since it is not needed to answer the question. (licensee 07/16)
- Added "as necessary" to answer D. (licensee 09/11)

Question: 2 Answer: B

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1 Pt(s) Given the following conditions:

- The plant is operating at 100% power.
- A Bus Differential Overcurrent condition occurs on 10A401.
- All equipment operates as designed.
- NO operator action is taken.

Which choice below describes the response of the AC Power Distribution to the conditions above?

A. Normal bus feeder breaker trips.
Alternate bus feeder breaker closes.
Motor load breakers remain closed.
USS feeder breakers remain closed.
EDG does NOT start

B. Normal bus feeder breaker trips.
Alternate bus feeder breaker remains open.
Motor load breakers trip.
USS feeder breakers trip.
EDG does NOT start

C. Normal bus feeder breaker trips.
Alternate bus feeder breaker remains open.
Motor load breakers trip.
USS feeder breakers remain closed.
EDG starts.
EDG output breaker closes

D. Normal bus feeder breaker trips.
Alternate bus feeder breaker trips.
Motor load breakers trip.
USS feeder breakers trip.
EDG starts.
EDG output breaker remains open

Distracter Analysis:

A. **Incorrect**: Alternate feeder breaker does not close. Motor load breakers trip. USS feeder breakers trip.
Plausible:

B. **Correct**

C. **Incorrect**: USS feeder breakers trip. The EDG regular Lockout Relay prevents EDG start and prevents EDG breaker closure.
Plausible:

D. **Incorrect**: The EDG regular Lockout Relay prevents EDG start.
Plausible:

Level: RO Exam

Lesson Plan Objective: 1EAC00E026

Source: Hope Creek Bank (Q56863)

Level of knowledge: memory

Reference(s): HC.OP-SO.PB-001 Section 3.3

KA: 295003.K2.03
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: (CFR: 41.7 / 45.8) AK2.03 A.C. electrical distribution system 3.7 3.9

Comments:

- Revised statement in the stem giving the condition that the bus tripped and modified to state that a bus overcurrent trip condition occurred. (Chief comment 07/16)

Question: 3 Answer: B

1 Pt(s) With the plant operating at rated power, the power supply fuse to a backup scram valve fails.

Which one of the following identifies the response of the associated backup scram valve and scram response due to this failure?

A. Valve repositions to trip position but NO scram occurs.

B. Valve CANNOT reposition but redundant valves can affect scram if an RPS trip occurs.

C. Valve CANNOT reposition and NO scram can occur even if an RPS trip occurs.

D. Valve repositions to trip position and a full scram occurs.

Distracter Analysis:

A. **Incorrect**: Valve needs to energize to trip (vent).
Plausible:

B. **Correct:** DC solenoid valve is energize to trip (vent). There are two redundant valves, and either one can perform the trip function.

C. **Incorrect**: First half is correct; however, a scram can still occur via the redundant valve.
Plausible:

D. **Incorrect**: Valve needs to energize to trip (vent) in order for a scram to occur.
Plausible:

Level: RO Exam

Lesson Plan Objective: RPS000E009

Source: Hope Creek Bank (Q68884)
Last NRC Exam 2002

Level of knowledge: Comprehension

Reference(s): HC.OP-SO.SB-0001

KA: 295003.K2.03

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : (CFR: 41.5 / 45.6) AK3.03 Reactor SCRAM: Plant-Specific 3.1 3.5

Comments:

- None

Question: 4 Answer: B

1 Pt(s) Given the following:

- The Unit is operating at 75% power.
- "A" EHC pump is out of service for maintenance.
- The Rx Operator observes EHC pressure at 1200 psig and dropping, AND the Operator immediately places the REACTOR MODE SWITCH in SHUTDOWN.
- EHC pressure stabilizes at about 800 psig within the next 5 seconds.
- Assume NO other operator actions.

Which ONE of the following describes the response of the electrical power system?

A. Generator output breakers BS2-6 AND BS6-5 automatically OPEN immediately upon placing the REACTOR MODE SWITCH in SHUTDOWN.

B. Generator output breakers BS2-6 AND BS6-5 automatically OPEN 2.5 seconds after Reverse Power Relay activation.

C. Generator output breakers BS2-6 AND BS6-5 automatically OPEN 30 seconds after Reverse Power Relay activation.

D. Generator output breakers BS2-6 AND BS6-5 must be MANUALLY OPENED in this situation because the Reverse Power feature is bypassed with the REACTOR MODE SWITCH in SHUTDOWN.

Distracter Analysis:

A. **Incorrect**: There is no direct tie between RPS logic and Main Turbine Generator or Output Breaker trip logic.
Plausible: A reactor scram will indirectly result in a generator lockout and output breaker trip on Reverse Power as the turbine coasts down.

B. **Correct**: A generator lockout will occur in 2.5 seconds after Reverse Power is sensed coincident with a turbine trip signal. In this case, a turbine trip signal came in at 1100 psig EHC pressure.

C. **Incorrect**: The 30 second time delay only comes into play if a turbine trip signal is not present. In this case, a turbine trip signal was present.

Plausible: A 30 second TD Reverse Power generator lockout does exist. The candidate must recognize the fact that it is only active when a turbine trip is not present.

D. **Incorrect**: The breakers will open automatically without operator action due to a Reverse Power generator lockout, and there is no mode switch bypass related to the Reverse Power protective feature.
Plausible: Procedures direct that the operators verify that a lockout has occurred at 0 MWe, and that is accomplished by manually opening the output breakers. In addition, the mode switch does bypass some interlocks, but reverse power is not one of them.

Level: RO Exam

Lesson Plan Objective: MNPWR0E019
(R) From memory list/identify the four (4) automatic actions which will occur if a generator lockout is initiated (exclude alarms).

Source: New

Level of knowledge: Comprehension

Reference(s):
NOH01MNTURB-03, "Main Turbine Construction & Components"
NOH01MNGENOC-05, "Main Generator System"
NOH01MNPWRO-06, "Main Power"
NOH01EHCOIL-06, "EHC Control Oil"

KA: 295005.A1.07
Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP : (CFR: 41.7 / 45.6) AA1.07 A.C. electrical distribution. 3.3 3.3

Comments:
- Simplified stem and used bullets. (Chief & licensee comments 07/16)

Question: 5 Answer: A

1 Pt(s) Given the following:

- The Unit is in OPCON 2 and a hot startup has just commenced.
- Reactor pressure is 250 psig.
- Bypass valves are controlling RPV pressure automatically.
- Main Turbine shell warming is in progress.
- The first control rod was just fully withdrawn.
- Turbine first stage pressure rises to 137 psig a few minutes later.
- The RWM Operator Display screen is on the CONFIRM SHUTDOWN screen, and is displaying the following:

ALL RODS IN:	NO
SHUTDOWN:	NO
RODS NOT FULL-IN:	001

Which ONE of the following describes the current plant status?

A. The withdrawn control rod failed to scram; however, the reactor is shutdown and will remain shutdown under all conditions.

B. Plant startup is proceeding normally and the current RWM indications are also normal when the reactor mode switch is in STARTUP with all rods NOT fully inserted.

C. The withdrawn control rod failed to scram and the reactor could go critical if a cool down is initiated.

D. The withdrawn control rod failed to scram and the reactor is critical.

Distracter Analysis:

A. **Correct**: A reactor scram signal (Main Turbine Stop Valves not full open with power greater than 30%) will be generated if Main Turbine shell warming is in service and 1^{st} stage pressure exceeds 135.7 psig. In shell warming, the stop valves are fully closed. 30% power is sensed when 1^{st} stage pressure reaches 135.7 psig. Turbine procedural cautions warn operators of this fact. So, one rod failed to scram in this case. In addition, the reactor will remain shutdown under all conditions without boron with the most reactive rod fully withdrawn (shutdown margin requirement). The RWM indications are correct for this situation. The RWM automatically switches to the CONFIRM SHUTDOWN screen when a scram is sensed. The SHUTDOWN = NO indication is due to the way the RWM is programmed. If any one rod is withdrawn beyond 02, then the NO flag will display. The RWM lesson plan highlights this human error trap. The operator is required to understand that the reactor is and will remain shutdown with one rod fully withdrawn regardless of the RWM indication.

B. **Incorrect**: The startup is NOT proceeding normally as described above, and if it was, then the RWM CONFIRM SHUTDOWN screen should NOT be displaying.
Plausible:

C. **Incorrect**: The control rod did fail to scram, but the reactor should remain shutdown under all conditions without boron as discussed in the justification for answer A.
Plausible:

D. **Incorrect**: The control rod did fail to scram; however, the reactor should NOT be able to go critical with just one rod fully withdrawn as discussed above.
Plausible:

Level: RO Exam

Lesson Plan Objective: ????

Source: New

Level of knowledge: Analysis

Reference(s):
Lesson Plan NOH04RODMINC, "Rod Worth Minimizer"
Lesson Plan NOH01MNTURB, "Main Turbine Construction and Components"
HC.OP-SO.AC-0001, "Main Turbine Operation"

KA: 295006.A2.05
Ability to determine and/or interpret the following as they apply to SCRAM: (CFR: 41.10 / 43.5 / 45.13) AA2.05 Whether a reactor SCRAM has occurred.4.6* 4.6*

Comments:

- Deleted "best" from the stem question. (licensee comment 07/16)

Question: 6 Answer: D

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1 Pt(s) The plant is operating at 100% power. Thick black smoke enters the Main Control Room requiring that the Control Room be abandoned. Plant equipment has NOT been affected by the condition. All Main Control Room actions, as specified in HC.OP-AB.HVAC-002(Q), "Control Room Environment", have been completed. NO other actions have been taken.

Which ONE of the following describes the plant parameter(s), if any, that are expected to exceed Emergency Operating Procedure entry conditions?

A. None

B. Reactor Level ONLY

C. Reactor Pressure ONLY

D. Reactor Level AND Reactor Pressure

Distracter Analysis:

A. **Incorrect**: See D below,
Plausible: If the control room is abandoned before any operator action is taken, then the plant will remain in steady state operation until control is established at the RSP. In addition, an integrated level of plant knowledge is required to understand the impact that closing the MSIV's has on feed water as well as reactor pressure control. Furthermore, knowledge of the procedure is required through training to know the control room actions taken prior to evacuation.

B. **Incorrect**: See D below.
Plausible: Reactor level will be affected as described in D below.

C. **Incorrect**: See D below.
Plausible: Reactor level will be affected as described in D below.

D. **Correct**: The procedure directs that the MSIV's be closed. This will result in a loss of the steam driven feed water pumps, and pressure will rise until SRV's lift since bypass valve capability is lost when the MSIV's are closed. Level will drop below 12.5 inches, and RPV Pressure will rise above 1037 psig, and these are both entry conditions to EOP-101 for RPV Control.

Level: RO Exam

Lesson Plan Objective: ABHVC2E004
Explain the reasons for how plant/system parameters respond when implementing Control Room Environment.

Source: New

Level of knowledge: Comprehension

Reference(s):
HC.OP-AB.HVAC.002(Q), "Control Room Environment"

KA: 295016.G2.4.4
Control Room abandonment. 2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (CFR 41.10 / 43.2 / 45.6)

Comments / Change Record:

- Capitalized NOT / NO & deleted "best" from the stem question. (licensee 07/16)

Question: 7 Answer: B

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1 Pt(s) Given the following plant conditions:

- The 'A' SACS Loop is in service supplying TACS.
- The 'D' SACS pump is running.
- The 'B' SACS Pump is Cleared and Tagged.
- The 'A' Fuel Pool Cooling Heat Exchanger (FPCC side only) is isolated for a piping leak repair.

A lightning strike results in an 'A' channel LOCA Level 1 signal and the loss of the 10A404 4KV bus.

What is the status of the Fuel Pool Cooling System for these conditions?

A. **Fuel Pool Cooling heat removal is unaffected by these conditions.**

B. **Fuel Pool Cooling heat removal has been lost and can be restored when the LOCA signal is reset.**

C. **Fuel Pool Cooling heat removal is being provided by the 'B' SACS Loop.**

D. **The Fuel Pool Cooling heat exchanger cross-tie valves auto open to provide Loop 'A' SACS flow to the 'B' FPCC heat exchanger.**

Distracter Analysis:

A. **Incorrect**: The loss of the 10A404 Bus results in the loss of the 'D' SACS Pump and the 'B' SACS loop and the loss of cooling to the only in service FPCC HX.
Plausible:

B. **Correct**: The loss of the 10A404 Bus results in the loss of the 'D' SACS Pump and the 'B' SACS loop. With the 'A' FPCC HX OOS on the FPCC side, all SACS cooling to FPCC is lost. The Cross-tie valve HV-2317A and HV-7922A receive close signals from the LOCA signal and cannot be opened without clearing the signal. The LOCA signal must be cleared to open the cross-tie valves and supply 'A' loop SACS flow to the 'B' FPCC HX.

C. **Incorrect**: The loss of the 10A404 Bus results in the loss of the 'B' SACS Loop.
Plausible:

D. **Incorrect**: The valves close on a LOCA signal and have no auto open signals.
Plausible:

Level: RO Exam

Lesson Plan Objective: ????

Source: HC Bank Q60630

Level of knowledge: Analysis

Reference(s):
HC.OP-SO.SM-0001

KA: 295018.K1.01
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.8 to 41.10) AK1.01 Effects on component/system operations 3.5 3.6

Comments / Change Record:
- None

Question: 8 Answer: C

1 Pt(s) A loose fitting has resulted in a loss of instrument air to the in-service Control Rod Drive (CRD) Flow Control Valve.

Determine which of the following conditions could result from this instrument air loss.

A. Control Rod Drive accumulator alarms due to low pressure.

B. High rod speeds during control rod withdraw.

C. Control Rod Drive alarms due to high temperatures.

D. Control Rods begin to drift due to excessive flow.

Distracter Analysis:

A. **Incorrect**: The charging header maintains pressure on the accumulators, and the charging header taps off upstream of the FCV, so accumulator pressure will not go down.
Plausible:

B. **Incorrect**: The drive header is downstream of the FCV, so rod withdraw speeds will be slower rather than faster.
Plausible:

C. **Correct:** The FCV fails to the minimum position on a loss of air. The cooling water header is downstream of the FCV. So, the failure mode results in decreased flow to the cooling water header and temperatures will rise resulting in CRD high temperature alarms.

D. **Incorrect**: The failure results in low flow. The failure mode does not cause an increased DP across the drive piston, so there is no motive force to cause the rods to drift.
Plausible:

Level: RO Exam

Lesson Plan Objective: ????

Source: INPO Bank (PB question)

Level of knowledge: Comprehension

Reference(s):
P&ID M-46
Lesson Plan NOH02000006C, "Control Rod Drive Hydraulics"

KA: 295019.K2.01
Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following:
(CFR: 41.7 / 45.8) AK2.01 CRD Hydraulics 3.8 3.9

Comments / Change Record:
- Corrected K/A from AK2.02 to AK2.01. (licensee comment 07/16)

Question: 9 Answer: C

1 Pt(s) Given the following:

- The Unit is in OPCON 4 with RCS temperature at 175°F
- "B" Loop RHR is in service.
- "A" Loop RHR is out of service for maintenance.
- "A" & "B" Reactor Recirculation Pumps are out of service for maintenance.
- RWCU is in service with 2 pumps and 2 filters.
- "B" RHR pump trips due to a motor failure.

Raising RWCU total system flow will.....

A. **increase core forced circulation inside the shroud.**

B. **prevent flashing in the RWCU suction flow venturi as the reactor heats up.**

C. **increase the reactor heat removal rate in the non-regenerative heat exchanger.**

D. **increase the reactor heat removal rate in the regenerative heat exchanger.**

Distracter Analysis:

A. **Incorrect**: Increasing RWCU flow will not increase core forced circulation inside the shroud.
Plausible: Increasing RWCU flow will affect RPV flow in the annulus and bottom head area.

B. **Incorrect**: Increasing flow will increase the possibility of flashing in the suction venturi.
Plausible: Suction venturi flashing is a concern as RPV temperature rises and it flashing is flow dependent as well, so someone could have the misconception that increasing flow will improve the margin to flashing.

C. **Correct:** Increasing RWCU system flow will increase the reactor heat removal rate in the non-regenerative heat exchanger.

D. **Incorrect**: The regenerative heat exchanger is not a heat sink; it is used for reheat. Therefore, increasing system flow will not assist with reactor heat removal. The regenerative heat exchanger is fully bypassed in order to maximize the reactor heat removal rate.

Plausible: The regenerative heat exchanger is a heat exchanger so a plausible misconception is that it will assist with reactor heat removal.

Level: RO Exam

Lesson Plan Objective: RWCU00E003
(R) Given the necessary sheets of P&ID's M-44 and M-45:
a. Determine the normal RWCU System flowpath IAW the RWCU System Lesson Plan.
b. Determine the blowdown RWCU System flowpath(s) IAW the RWCU System Lesson Plan.
c. Determine the recirculation RWCU System flowpath IAW the RWCU System Lesson Plan.

Source: New

Level of knowledge: Comprehension

Reference(s):
HC.OP-AB.RPV-0009, "Shutdown Cooling"

KA: 295021.K3.04
Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING: (CFR: 41.5 / 45.6) AK3.04 Maximizing reactor water cleanup flow.3.3 3.4

Comments / Change Record:

- Added the words "inside the shroud" to answer "A". (licensee comment 09/11)

Question: 10 Answer: A

1 Pt(s) Given the following:

- The Unit is in OPCON 5.
- Reactor cavity gates installed.
- Cavity drain down begins in preparation for reactor reassembly.
- Cavity level is lowered to the top of the reactor flange.
- 30 minutes later, operators report that Fuel Pool level is visibly dropping due to a significant leak through the cavity gate seals.
- Radiation level rises from 15 to 25 mr/hr at the Fuel Storage Pool area radiation monitor 9RX707.
- Operators are able to maintain Fuel Pool level above the low level alarm set point.

Which one of the following is the correct EAL declaration?

A. An EAL declaration is NOT required.

B. Declare an Unusual Event as a result of lowering reactor cavity level.

C. Declare an Unusual Event as a result of lowering Fuel Pool level.

D. Declare an Alert due to the potential to uncover irradiated fuel.

Distracter Analysis:

A. **Correct:** An EAL threshold has not been reached, so no declaration is required.

B. **Incorrect**: The reactor cavity level was lowered as part of a planned and controlled evolution; therefore, the EAL is not met.
Plausible: The RX cavity letdown was planned, but the fuel pool level drop was not. Need to understand the difference between planned and unplanned relative to the EAL.

C. **Incorrect**: The fuel pool level was not intentionally lowered; however, the operators were able to control the level reduction since level is being maintained above the alarm set point. So, the EAL is not met.
Plausible:

D. **Incorrect**: Radiation levels are well below the EAL threshold, and the current levels in the fuel pool and reactor cavity present no significant risk of uncovering irradiated fuel. The EAL is not met.
Plausible:

Level: RO Exam

Lesson Plan Objective: ????

Source: New

Level of knowledge: Analysis

Reference(s):
Hope Creek EALs & Bases document

KA: 295023.K2.05
Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: (CFR: 41.7 / 45.6) AA2.05 †Entry conditions of emergency plan 3.2 4.6*

Comments / Change Record:

- Changed 95X707 to 95R707. (licensee comment 07/16)
- Changed radiation levels from "rise to a high of 210 mr/hr" to rise from 15 to 25 mr/hr. (licensee comment 07/16)
- Corrected typo in stem question. (licensee comment 07/16)
- Capitalized "Fuel Pool". (licensee comment 08/27)
- Licensee requested that this question be classified as SRO only; however, the request was **denied** based on NUREG-1021 guidance. This is a valid K/A with RO importance value greater than 2.5 (value is 3.2). (licensee comment)

Question: 11 Answer: C

1 Pt(s) A transient has occurred. The current containment conditions are as follows:

- Drywell pressure: 4.5 psig rising
- Drywell temperature: 150 F rising
- Torus pressure: 4.0 psig rising
- Torus water temperature: 82 F steady
- Torus airspace temperature: 145 F rising

Which of the following events would explain the current plant conditions?

A. **A safety relief valve (SRV) has lifted and is discharging through a T-Quencher.**

B. **The containment is functioning normally following a feedwater line break inside containment.**

C. **A main steam line break has occurred inside containment with a torus to drywell vacuum breaker open.**

D. **The containment is functioning normally following a total loss of drywell cooling.**

Distracter Analysis:

A. **Incorrect**: When an SRV lifts, then it normally discharges below the suppression pool water level through a T-Quencher. If this were the event in progress, then torus water temperature would be noticeably rising with a minimal increase in torus airspace temperature and pressure since the steam would be quenched by the pool. Drywell pressure and temperatures should remain normal.
Plausible:

B. **Incorrect**: Drywell conditions would look as indicated; however, the torus indications are wrong for this event (feedwater line break). For this event, torus water temperature should be rising as steam is forced down through the LOCA downcomers and into the pool water. Drywell pressure should be about 1.5 psig higher than torus pressure as the drywell pressure displaces the normal water column in the downcomer to push the steam into the pool. The given indications suggest that normal pressure suppression flow path has been compromised, and one of the more likely bypass paths would be through a stuck open or leaking torus to drywell vacuum breaker.

Plausible:

C. **Correct:** The drywell and torus airspace conditions are equalizing, but the suppression pool water temperature is normal and not rising. The condition is indicative of a high energy line break inside the drywell with a suppression pool bypass flow path, so a main steam line break inside the drywell with a leaking torus to drywell vacuum breaker could yield the given indications.

D. **Incorrect**: Drywell conditions could possibly be explained by a total loss of cooling over a long period of time; however, the given drywell pressure should not be reached within 10 minutes of a loss of cooling event. In addition, the torus conditions indicate a suppression pool bypass flow path due to the low delta P between the torus and drywell.
Plausible:

Level: RO Exam

Lesson Plan Objective: PRICONE008
(R) Predict the possible consequences of the following conditions for the Torus-to-Drywell Vacuum Breakers and Rx. Bldg.-to-Torus Vacuum Breakers during a response to a LOCA:

a. Failed Open
b. Failed Closed

Source: Mod INPO Bank PB2 Question

Level of knowledge: Analysis

Reference(s):
NOH01PRICONC, "Primary Containment Structure"

KA: 295024.A2.05
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL PRESSURE: (CFR: 41.10 / 43.5 / 45.13) EA2.05 Suppression chamber air-space temperature: 3.6 3.7

Comments / Change Record:

- Changed "most likely" to "would" in the stem question. (licensee comment 07/16)
- Changed "leaking" to "open" in answer C. (licensee comment 07/16)
- Changed "A transient occurred 10 minutes ago" to "A transient occurred". (Chief comment 07/16)

Question: 12 Answer: B

1 Pt(s) Given the following:

- The Unit is in OPCON 4.
- RPV bulk coolant temperature is 110 °F.
- All RPV metal temperatures are 110 ± 5 °F.
- The Outage Control Center has requested that RPV pressure be raised to RATED pressure for hydrostatic testing while at the current RPV temperatures.

Which one of the following is the specific reason why the request should be denied?

A. **The core shroud could crack due to pressurized thermal shock.**

B. **The RPV could experience brittle fracture.**

C. **SRV's could prematurely lift since the setpoints are based on hot calibration conditions.**

D. **The RPV pressure Safety Limit for OPCON 4 would be exceeded.**

Distracter Analysis:

A. **Incorrect**: Core shroud cracking is a BWR concern, but the failure mechanism is not related to pressurized thermal shock. Core shroud cracking is believed to be due to stress corrosion cracking. Pressurized thermal shock is more applicable to PWR's and it occurs when a component with a high delta P across it experiences a high rate temperature change. The core shroud is a vessel internal component, and the component is not subjected to any significant delta P conditions during hydrostatic testing. Furthermore, the given condition holds temperature constant.
Plausible:

B. **Correct:** The bases for Tech Spec 3.4.6 clearly discuss brittle fracture as the primary concern. Tech Spec 3.4.6.1 and Tech Spec Figure 3.4.6.1-1 depict the limits.

C. **Incorrect**: The bases for Tech Spec 3.4.6 clearly discuss brittle fracture as the primary concern. SRV lift setpoint may vary as a function of temperature conditions, but even if it does, the brittle fracture issue is the specific reason why Tech Specs forbids raising

pressure to rated with the given temperature conditions, and that is specific reason why an SRO would deny such a request.
Plausible:

D. **Incorrect**: The RPV Pressure Safety Limit is 1325 psig and is applicable in OPCONS 1, 2, 3, & 4. This limit will not be approached if pressure is raised to rated conditions.
Plausible:

Level: RO Exam

Lesson Plan Objective: RXVESSE009
(R) Given plant problems/industry events associated with the Reactor Vessel and Internals:

a. Discuss the root cause of the plant problem/industry event IAW the plant/industry event.
b. Discuss the HCGS design and/or procedural guidelines that mitigate/reduce the likelihood of the problem/industry event at HCGS IAW the plant/ industry event.
c. Discuss the "lessons learned" from this problem/event IAW the plant/industry event.

Source: New

Level of knowledge: Comprehension

Reference(s):
Tech Spec 3.4.6 & Bases
NOH01RXVESSC, "Reactor Vessel and Internals"

KA: 295025.G2.1.32
2.1.32 Ability to explain and apply all system limits and precautions.
(CFR: 41.10 / 43.2 / 45.12)

Comments / Change Record:

- Changed stem to remove reference to the SRO, and deleted the word "between" before the temperature range in the third bullet of the stem. (licensee comment 07/16)
- Capitalized "RATED". (licensee comment 09/11)

Question: 13 Answer: C

1 Pt(s) The reactor has scrammed on high drywell pressure. Plant conditions are as follows:

- Reactor pressure: 50 psig, stable
- RPV water level: -50 inches, rising slowly
- Suppression pool temperature: 200 °F, rising slowly
- Suppression chamber pressure: 10 psig, stable
- Suppression pool water level: 0 inches, stable
- "B" RHR is in Torus Cooling and Spray: 10,000 gpm

For the conditions stated, which of the following is the highest suppression pool temperature that ensures NPSH is maintained for the "B" RHR pump?

A. **206 °F (± 1 °F)**

B. **223 °F (± 1 °F)**

C. **233 °F (± 1 °F)**

D. **246 °F (± 1 °F)**

Distracter Analysis:

A. **Incorrect**: Based on overpressures of 0 psig.
Plausible:
B. **Incorrect**: Based on overpressures of 5 psig.
Plausible:
C. **Correct:** EOP Caution 2 based on 10 psig suppression chamber overpressure curve.
D. **Incorrect**: Based on overpressures of 15 psig.
Plausible:

Level: RO Exam

Lesson Plan Objective: EOP102E009
(R) Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Suppression Pool Lesson Plan.

Source: HC Bank Q71323

Level of knowledge: Comprehension

Reference(s):
EOP Caution 2

KA: 295026.K1.01
Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: (CFR: 41.8 to 41.10) EK1.01 Pump NPSH 3.0 3.4

Question: 14 Answer: D

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1 Pt(s) Given the following conditions:

- The plant is at 100% power.
- Digital feedwater control is in MANUAL.
- A loss of drywell cooling causes elevated drywell temperature and pressure.

Which statement below describes Narrow Range Level Instrument response and the reason for this response? Assume NO operator action.

A. There will be NO change in indicated level because the reference and variable leg densities both increase.

B. There will be NO change in indicated level because narrow range instruments are density compensated.

C. Indicated level will decrease because of increased reference leg density.

D. Indicated level will increase because of decreased reference leg density.

Distracter Analysis:

A. **Incorrect**: Both densities will decrease, but the reference leg has more vertical piping that is subject to the high temperature conditions.
Plausible:

B. **Incorrect**: The reference leg has more vertical piping that is subject to the high temperature condition thereby causing indicated level to rise.
Plausible:

C. **Incorrect**: Rising drywell temperature will lower reference leg density causing indicated level to rise.
Plausible:

D. **Correct:** Level detectors measure the DP between the reference leg (Hi press) and the variable leg (Low press). With actual RPV level high, the Dp is at its smallest value. Any condition causing the pressure exerted by the reference leg to drop or cause the pressure exerted by the variable leg to rise will cause indicated level to rise or be higher than actual. The converse also applies.

Level: RO Exam

Lesson Plan Objective:
RXINSTE018
INSTREE001

Source: HC Bank Q54798
Last used on 9/13/99

Level of knowledge: Comprehension

Reference(s):

KA: 295028.K2.02
Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: (CFR: 41.7 / 45.8) EK2.02 Components internal to the drywell 3.2 3.3

Comments / Change Record:

- Rephrased all answers to provide a parallel sentence structure for improved psychometric balance. (licensee comment 07/16)
- Added a bullet in the stem to state "Digital feedwater control is in MANUAL" and added "Assume NO operator action" after the stem question. These conditions are needed for "D" to be the correct answer. (licensee comment 09/11)

Question: 15 Answer: B

1 Pt(s) Given the following:

- The reactor is at 100% power.
- A suppression pool level transient is in progress.
- Suppression pool level reaches 54 inches and continues to lower.
- A manual reactor scram is inserted in accordance with the EOP's.

What is the basis for this action?

A. **The SRV tailpipe vacuum breakers are uncovered, so the containment will be directly pressurized if the SRV's lift.**

B. **The torus vent header drain lines are almost uncovered, so the containment could be directly pressurized in the event of a LOCA as pool level continues to lower.**

C. **Tech Specs require an immediate reactor scram.**

D. **Low Pressure ECCS is inoperable and unavailable due to NPSH and vortex concerns.**

Distracter Analysis:

A. **Incorrect**: The SRV tailpipe vacuum breakers are located in the drywell and are supposed to be uncovered to allow proper operation once an SRV closes and steam in the tailpipe condenses. Also, SRV lift should not cause the vacuum breaker to open while the SRV is actively discharging.
Plausible: SRV tailpipe vacuum breakers are normally uncovered. A failed open or leaking vacuum breaker could result in direct containment pressurization upon SRV lift, but the stem of the question does not suggest that there are any problems with the vacuum breaker. The distracter tests the candidates' knowledge of the location and operation of the vacuum breaker.

B. **Correct**: The EOP Basis document provides several reasons for directing a manual scram as a prudent anticipatory action. The most immediate reason is that if suppression pool level cannot be maintained above 55 inches, then the torus vent header 1 ¼ inch drain lines are close to becoming uncovered (uncovered at 50 inches). These drain lines are not capped or valved. If these lines are uncovered, then the SP airspace could be directly pressurized in the event of a LOCA since some steam would not be quenched by the pool.

C. **Incorrect**: LCO 3.5.3.a Action A requires that level be restored to at least 74.5 inches within 1 hour or be in hot shutdown within the next 12 hours. An immediate scram is not required for this condition.
Plausible: The EOP bases state that that a manual scram is a prudent action to take since efforts to restore level above the LCO requirement have been unsuccessful and that further level degradation is likely to cause the torus vent drain lines to become uncovered.

D. **Incorrect**: Low pressure ECCS remains available despite *potential* operability concerns related to NPSH & vortexing. In addition, the EOP bases do NOT list this as a basis to initiate a manual scram.
Plausible: The Tech Spec LCO 3.5.3 bases mention that NPSH and vortex issues are a concern as level drops below 74.5 inches, but that is not the reason that the EOP's direct a manual scram.

Level: RO Exam

Lesson Plan Objective: PRICONE003
Summarize the basic construction and function of the following Primary Containment components:

- Vent header
- Etc.

Source: New

Level of knowledge: Memory

Reference(s):
Tech Spec 5.5.3 & Bases
EOP 102 & Bases

KA: 295030.K3.06
Knowledge of the reasons for the following responses as they apply to LOW SUPPRESSION POOL WATER LEVEL: (CFR: 41.5 / 45.6) EK3.06 Reactor SCRAM. 3.6 3.8

Comments / Change Record:

- Changed stem to provide an initial reactor power. (NRC reviewer comment 07/16)
- Added words to answer "B": ",so the containment could be directly pressurized in the event of a LOCA as pool level continues to lower". (licensee comment 07/16)

Question: 16 Answer: D

1 Pt(s) Given the following:

- A small break LOCA is in progress.
- All control rods are fully inserted.
- All injection sources have failed.
- Reactor level is – 170 inches and going down slowly.
- Reactor pressure is 900 psig and trending down very slowly.
- Operators just began to perform OP-EO.ZZ-310, "Alternate Injection Using Fire Water", but those actions are NOT yet complete.

When must the operators open 5 ADS valves?

A. BEFORE RPV level drops to – 185 inches.

B. WHEN RPV level CANNOT be restored AND maintained above – 185 inches.

C. BEFORE RPV level drops to – 200 inches.

D. WHEN RPV level drops to – 200 inches.

Distracter Analysis:

A. **Incorrect**: An emergency blowdown is allowed by EOP-101 before – 185 is reached if sufficient injection sources are available to restore level above – 129 inches or restore and maintain it above – 185 in following the blowdown. In the scenario given, there is zero injection available.
Plausible:

B. **Incorrect**: If level cannot be restored and maintained above – 185 inches with an injection source available, then SAG entry is required for containment flooding. With no injection lined up and ready to inject, EOP-101 directs the use of steam cooling down to -200 inches before an emergency blowdown is directed.
Plausible:

C. **Incorrect**: See explanation for D. With zero injection capability, it is not appropriate to blowdown until – 200 inches is reached since steam cooling should be used as long as possible until it is no longer effective to maintain adequate core cooling.
Plausible:

D. **Correct:** With zero injection, EOP-101 directs that the operators wait until the Minimum Zero Injection Water Level is reached (MZIRWL, - 200 inches). This allows steam cooling to be used for as long as possible before an emergency depressurization is performed. Once MZIRWL is reached, then adequate core cooling is not assured and an emergency depressurization is required even without an injection source available.

Level: RO Exam

Lesson Plan Objective: EO101PE008
(R) Given any step of the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by that step.

Source: New

Level of knowledge: Analysis

Reference(s):
EOP-101 and EOP-202

KA: 295031.A1.07
Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL: (CFR: 41.7 / 45.6) EA1.07 Safety/relief valves 3.7* 3.7*

Comments / Change Record:

- Capitalized NOT in the last stem bullet. (licensee comment 07/16)

Question: 17 Answer: C

1 Pt(s) The Unit was operating at 100% power when the following event occurred:

- The reactor scrammed due to high reactor pressure.
- Reactor pressure peaked at 1115 psig during the transient, and pressure is now at 900 psig and lowering slowly.
- 5 control rods still indicate position 48.
- All APRM downscale lights are extinguished.
- One SRV is stuck open.
- Suppression pool temperature is 112 °F and rising.
- Reactor level is at – 25 inches and lowering slowly.
- Drywell pressure is 0.5 psig and stable.

Which ONE of the following is the correct operator action to take based on these conditions?

A. Maintain RPV level between - 25 inches and + 54 inches.

B. Lower RPV level until UNTIL level drops below – 50 inches.

C. Lower RPV level UNTIL reactor power drops below 4% OR RPV level reaches – 129 inches OR all SRV's are closed.

D. Maintain RPV level between -185 inches and – 25 inches.

Distracter Analysis:

A. **Incorrect**: With an ATWS in progress with power above 4%, maintaining the current reactor level or allowing it to rise will allow energy addition to the containment to continue. In fact, raising level will tend to raise power making the situation worse.
Plausible:

B. **Incorrect**: This answer would be correct if suppression pool temperature was less than 100 °F, but it is not, so answer C is the correct action to take.
Plausible:

C. **Correct:** Answer C is correct since reactor power is above 4% (APRM downscale lights extinguished) and suppression pool temperature is above 110 °F with an SRV open.

D. **Incorrect**: Additional level reduction is required to either get power less than 4% or the SRV closed to help minimize containment energy addition. Once that is accomplished, then a maintenance

band is appropriate between – 185 inches and the level needed to keep power less than 4 % or to get the SRV to close or – 129 inches (if power does not drop to less than 4% or the SRV does not close).
Plausible:

Level: RO Exam

Lesson Plan Objective: EO101AE008
(R) Given any step of the procedure, explain the reason for performance of that step and/or evaluate the expected system response to control manipulations prescribed by that step.

Source: New

Level of knowledge: Analysis

Reference(s):
EOP-101A & bases

KA: 295037.A2.04
Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : (CFR: 41.10 / 43.5 / 45.13) EA2.04 Suppression pool temperature 4.0* 4.1*

Comments / Change Record:

- None

Question: 18 Answer: D

1 Pt(s) Given the following:

- A main steam line break has occurred in the turbine building immediately upstream of the main turbine stop valves.
- The break CANNOT be isolated.
- Steam is discharging into the turbine building.
- Radiation Protection reports that radiation levels are rising in the turbine building as well as outside the turbine building.
- The Emergency Coordinator just declared a Site Area Emergency.

Who should be notified as the first priority?

A. **The states and counties.**

B. **The NRC Operations Center.**

C. **The Site Vice President or designee.**

D. **On site plant personnel.**

Distracter Analysis:

A. **Incorrect**: The states & counties must be notified within the next 15 minutes to comply with regulations and ECG Attachment 6. This would be the second highest priority notification to make. The most immediate priority is to protect the health & safety of people that are on site, so that notification should be made first.
Plausible:

B. **Incorrect**: The NRC Ops Center must be notified within 60 minutes, so this is not the first notification that needs to be made.
Plausible:

C. **Incorrect**: There is no time requirement to make this notification even though it may be an important notification to make. It may be made as soon as practical and it is not the first priority.
Plausible:

D. **Correct:** A plant page notification should be made immediately to warn plant personnel of the ongoing hazard. This action is required to protect the health and safety of people, and it is the number 1 priority. Other notifications still need to be made within specified time requirements, but this is the most immediate concern and notification should not be delayed.

Level: RO Exam

Lesson Plan Objective: ????

Source: New

Level of knowledge: Memory

Reference(s):
ECG Attachment 6
OP-AA-104-101, "Communications"

KA: 295038.G2.1.14
2.1.14 Knowledge of system status criteria which require the notification of plant personnel. (CFR: 43.5 / 45.12)

Comments / Change Record:

- Replaced "Health Physics" with "Radiation Protection"
- Replaced the word "Director" with "Coordinator". (licensee 09/11)

Question: 19 Answer: A

1 Pt(s) An electrical fire has been reported in the "A" Emergency Diesel Generator (EDG) room. The automatic fire suppression system actuated as designed. Fire response personnel are planning to enter the room to confirm that the fire is extinguished.

Which one of the following describes the fire class and the potential hazards associated with entering the room?

A. **Class C; suffocation hazard due to CO2 discharge.**

B. **Class C; electrocution hazard due to water deluge.**

C. **Class A; suffocation hazard due to CO2 discharge.**

D. **Class B; electrocution hazard due to water deluge.**

Distracter Analysis:

A. **Correct:** The fire is electrical in nature, so it is Class C. The EDG room is protected by automatic CO2 suppression, so suffocation is a major hazard to consider prior to entry.

B. **Incorrect**: Class C is correct; however, the room is NOT protected by water deluge.
Plausible:

C. **Incorrect**: Class A pertains to paper/wood fires and this fire was electrical in nature. The second part of the answer is correct.
Plausible:

D. **Incorrect**: Class B pertains to combustible liquids and this fire was electrical. In addition, the room is NOT protected by water deluge.
Plausible:

Level: RO Exam

Lesson Plan Objective: ????

Source: INPO Bank (Dresden 2)

Level of knowledge: Memory

Reference(s):
NOH04EDG000C, "Emergency Diesel Generator"

NOH01FIREPRO, "Fire Protection System"

KA: 600000.K1.01
Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: AK1.01 Fire Classifications by type 2.5 2.8

Comments / Change Record:

- Changed answer "C" from class B to class A to avoid the possibility that someone could assert that oil and fuel would eventually making "C" a potentially correct answer. (NRC exam reviewer 07/16)

Question: 20 Answer: A

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1 Pt(s) The Unit is in OPCON 1 with irradiated fuel moves in progress on the refueling floor. Ventilation systems are lined up in the normal lineup to support the plant conditions. The following event occurs:

- An irradiated fuel bundle is dropped from the full up position.
- The bundle impacts other irradiated bundles in the fuel pool.
- Bubbles are observed rising from the dropped bundle.
- Refuel Floor Exhaust radiation rises to 1.8 X 10E-3 μCi/cc.
- Reactor Building Exhaust radiation is normal.
- 60 seconds later the Reactor Operator checks ventilation status.

Which of the following describes secondary containment ventilation status?

A. **Reactor Building ventilation supply and exhaust fans are running, FRVS vent fans are NOT running, and FRVS recirculation fans are NOT running.**

B. **Reactor Building ventilation supply and exhaust fans are NOT running, ONE FRVS vent fan is running, and SIX FRVS recirculation fans are running.**

C. **Reactor Building ventilation supply and exhaust fans are NOT running, TWO FRVS vent fan are running, and SIX FRVS recirculation fans are running.**

D. **Reactor Building ventilation supply and exhaust fans are running, FRVS vent fans are NOT running, and FOUR FRVS recirculation fans are running.**

Distracter Analysis:

A. **Correct:** The Refuel Floor Exhaust high radiation isolation signal occurs at 2.0 X 10E-3 μCi/cc. In the given scenario, Refuel Floor Exhaust radiation levels only reached 1.8 X 10E-3 μCi/cc; therefore, plant ventilation system status remains unaffected. RBVS supply & exhaust fans are still running, and FRVS fans are not running.

B. **Incorrect**: This answer would be correct if the radiation levels exceeded the isolation setpoint, but the radiation levels are below the setpoint in this scenario, so this answer is not correct.
Plausible:

C. **Incorrect**: Answer C is partially correct when the radiation level exceeds the setpoint; however, only the lead FRVS vent fan would auto start on the initiation signal. The second FRVS vent fan will only automatically start if there is an initiation signal and the lead fan has low flow for 45 seconds.
Plausible:

D. **Incorrect**: Answer D is partially correct for a normal ventilation lineup; however, the FRVS recirculation fans are not part of the normal ventilation lineup.
Plausible:

Level: RO Exam

Lesson Plan Objective: SECCONE008
(R) Given a list of plant conditions, select the four automatic signals which will shutdown and isolate normal Reactor Building Ventilation and start the Filtration Recirculation and Ventilation System (FRVS).

Source: New

Level of knowledge: Memory

Reference(s):
HC.OP-AB.CONT-0005, "Irradiated Fuel Damage"
Lesson Plan NOH01RBVENTC, "Reactor Building Ventilation"

KA: 295023.K2.05
Knowledge of the interrelations between REFUELING ACCIDENTS and the following: (CFR: 41.7 / 45.8) AK2.05 Secondary containment ventilation 3.5 3.7

Comments / Change Record:

- None

Question: 21 Answer: C

1 Pt(s) The Unit was operating at 100% power when a total loss of feedwater occurred. HPCI and RCIC automatically started and RPV level rose to +56 inches.

Which of the following describes the correct valve status and operation?

A. **HV-4282, "RCIC Turbine Trip and Throttle Valve" is closed. The valve can ONLY be opened after the high level condition clears AND the operator manually resets the valve trip latch by using the control room close pushbutton to reengage the valve operator to the valve stem. Once this is done, then the operator can open the valve using the open pushbutton.**

B. **HV-4282, "RCIC Turbine Trip and Throttle Valve" is closed. If RPV level drops to – 38 inches, then the valve will open automatically without operator action.**

C. **HV-F045, "RCIC Turbine Steam Supply Isolation Valve" is closed. If RPV level drops to – 38 inches, then the valve will open automatically without operator action.**

D. **HV-F045, "RCIC Turbine Steam Supply Isolation Valve" is closed. The valve can ONLY be opened after the high level condition clears AND the operator manually resets the valve trip latch by using the control room close pushbutton to reengage the valve operator to the valve stem. Once this is done, then the operator can open the valve using the open pushbutton.**

Distracter Analysis:

A. **Incorrect**: HV-4282 does not trip on a high level shutdown signal. The operator action described is correct to reset the valve if it tripped; however, the valve should not have tripped in the given situation.
Plausible:

B. **Incorrect**: HV-4282 does not trip on a high level shutdown signal.
Plausible:

C. **Correct:** The valve closes on a +54 inch signal. The valve will automatically reopen on -38 inches. The system is designed this way to allow the system to cycle between -38 inches and + 54 inches without operator intervention.

D. **Incorrect**: The valve status is correct, but the operation describes the reset procedure for the HV-4282 valve.
Plausible:

Level: RO Exam

Lesson Plan Objective: RCIC00E005
(R) Given plant conditions regarding the RCIC trip and throttle valve:

a. Distinguish a mechanical overspeed trip from other trips IAW the RCIC System Lesson Plan.

b. Explain how to reset a mechanical overspeed trip IAW the system operating procedure.

c. Explain how the valve closes on a trip signal IAW the RCIC System Lesson Plan.

Source: New

Level of knowledge: Memory

Reference(s):
Lesson Plan HOH0RCIC00, "RCIC System"

KA: 295008.K3.08
Knowledge of the reasons for the following responses as they apply to HIGH REACTOR WATER LEVEL: (CFR: 41.5 / 45.6) AK3.08 RCIC steam supply valve closure: Plant-Specific.3.4 3.5

Comments / Change Record

- None

Question: 22 Answer: D

1 Pt(s) Given the following:

- The Unit is operating at 100% power.
- A feedwater malfunction causes 2 Reactor Feed Pumps to trip and the remaining feed pump to run down to minimum speed.
- RPV level briefly dipped below – 38 inches for about 5 seconds before operators were able to reestablish feed flow.
- RPV is now – 25 inches and rising slowly.
- All control rods fully inserted as required.

What is the status of the Reactor Recirculation Pumps 30 seconds after -38 inches was reached?

A. Both pumps are running at the pre-transient speed.

B. Both pumps are running at 45% speed.

C. Both pumps are running at 30% speed.

D. Both pumps are tripped.

Distracter Analysis:

A. **Incorrect**: Both pumps tripped 9 seconds after – 38 inches was reached.

B. **Incorrect**: The 45% speed limiter was activated; however, both pumps tripped 9 seconds after – 38 inches was reached.

C. **Incorrect:** The 30% speed limiter was activated as level dropped below + 12.5 inches during the transient; however, both pumps tripped 9 seconds after – 38 inches was reached.

D. **Correct**: Both pumps trip 9 seconds after – 38 inches is reached. The signal seals in, so the trip still occurs even if level recovers within the 9 second time delay.

Level: RO Exam

Lesson Plan Objective: RRCS00E003
For each of the four actuation logic circuits within the Redundant Reactivity Control System, from memory:
a. Explain how each functions to control reactivity.

b. Choose the setpoint(s) for each actuation and explain any associated permissive(s) required for the actuation to occur.

Source: New

Level of knowledge: Analysis

Reference(s):
Lesson Plan NOH01RECCON, "Reactor Recirc Flow Control"
Lesson Plan NOH01RECIRC, "Reactor Recirculation System"

KA: 295009.A1.03
Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL: (CFR: 41.7 / 45.6) AA1.03 Recirculation system: Plant-Specific. 3.0 3.1

Comments / Change Record:

- Replaced "30 seconds later" with "30 seconds after -38 inches was reached". (licensee comment 07/16)

Question: 23 Answer: D

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1 Pt(s) Given the following conditions:

- The plant is operating at 100% power.
- The COMPUTER PT IN ALARM A4-F5 alarm is received.
- Drywell pressure is 1.1 psig.
- HC.OP-AB.CONT-001, Drywell Pressure abnormal is entered.

Which of the following events, BY ITSELF, could be the cause of the pressure rise?

A. Failure of the "A" Reactor Recirculation Pump #2 Seal

B. FV-4971 Nitrogen Flow Control Valve fails open

C. Torus Vent Valve Isolation Valve HV-11541 fails open

D. Loss of power to multiple Drywell Fans

Distracter Analysis:

A. **Incorrect**: Assuming the #1 seal is intact, no change in DW conditions will occur.
Plausible:

B. **Incorrect**: During normal operation, the nitrogen FCV is isolated from the DW.
Plausible:

C. **Incorrect**: A vent valve opening would result in a reduction in pressure. However the rupture disk downstream of HV-11541 should be intact resulting in no effect on DW pressure.
Plausible:

D. **Correct:** A reduction in cooling will raise temperature and therefore pressure in the Drywell.

Level: RO Exam

Lesson Plan Objective:
ABCNT1E004

Source: HC Bank Q61761

Level of knowledge: Comprehension

Reference(s):
HC.OP-AB.CONT-0001

KA: 295012.A2.02
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13)
AA2.02 Drywell pressure. 3.9 4.1

Comments / Change Record:

- Changed DW pressure from 1.11 to 1.1 and capitalized "BY ITSELF". (licensee comment 07/16)
- Replaced "DRYWELL PRESSURE HI/LO" with "COMPUTER PT IN ALARM A4-F5". (licensee comment 09/11)

Question: 24 Answer: C

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1 Pt(s) Consider each of the following scenarios independently:

1. A Site Area Emergency was declared based on high FRVS release rate indications exceeding the EAL threshold.
2. An Alert was declared based on high SPV release rate indications exceeding the EAL threshold.
3. An Alert was declared based on a large radioactive liquid release to the river.

Select the answer below that provides a complete list of the scenarios which satisfy the entry conditions for HC.OP-EO.ZZ-0103/4, "Reactor Building & Rad Release Control".

A. **1 and 2 and 3**

B. **2 and 3 only**

C. **1 and 2 only**

D. **1 only**

Distracter Analysis:

A. **Incorrect**: EOP-104 is entered based on any **gaseous** radioactive release above the **Alert** EAL threshold. Scenario 3 deals with a **liquid** effluent release.
Plausible: Scenarios 1 & 2 do meet the entry conditions for EOP-104. Scenario 3 partially meets the entry conditions since it is a radioactive effluent release, there is an Alert EAL for this type of release, and finally, the EOP is entered at the Alert EAL level. However, Scenario 3 is NOT for a gaseous effluent release.

B. **Incorrect**: EOP-104 is entered based on any **gaseous** radioactive release above the **Alert** EAL threshold. Scenario 3 deals with a liquid effluent release.
Plausible: Scenario 2 does meet the entry conditions. Scenario 3 partially meets the entry conditions since it is a radioactive effluent release, there is an Alert EAL for this type of release, and finally, the EOP is entered at the Alert EAL level. However, Scenario 3 is NOT for a gaseous effluent release.

C. **Correct:** EOP-104 is entered based on any **gaseous** radioactive release above the **Alert** EAL threshold. Scenarios 1 AND 2 satisfies the entry requirements.

D. **Incorrect**: Scenario 2 is also a valid entry condition for EOP-104, so Scenario 1 alone is not correct.
Plausible: The Site Area Emergency EAL threshold could be mistaken as the EOP-104 entry threshold, and if it was, then the answer would be correct since Scenario 2 would be eliminated.

Level: RO Exam

Lesson Plan Objective: EOP103E002
Given a set of plant conditions, analyze and determine if entry conditions into HC.OP-EO.ZZ-0103/4 exists.

Source: New

Level of knowledge: Comprehension

Reference(s):
HC.OP-EO.ZZ-0103/4 & Bases

KA: 295017.G2.4.4
2.4.4 Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (CFR 41.10 / 43.2 / 45.6)

Comments / Change Record:

- Added "only" to answers B & C. (licensee comment 07/16)
- Changed procedure number from HC.OP-EO.ZZ-0103 to HC.OP-EO.ZZ-0103/4. (licensee comment 08/27)
- Licensee requested that this question be classified as SRO only; however, the request was **denied** since it deals with knowledge of EOP entry conditions. This K/A has an RO importance of 4.0, so it is a valid RO question. (licensee 09/11)
- Reclassified cognitive level from memory to comprehension since the question tests knowledge of the EAL threshold requiring EOP entry as well as the understanding that the procedure only applies to gaseous radioactive releases. (NRC Exam Author 09/14)

Question: 25 Answer: C

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1 Pt(s) Given the following:

- The Unit is critical in OPCON 2.
- Reactor pressure is 550 psig.
- "A" CRD pump is out of service for maintenance.
- "B" CRD pump tripped at 10:00 EDT due to a motor fault.
- At 10:05 EDT, at least 5 different CRD accumulator trouble alarms were received due to low accumulator pressure. The associated control rods are fully withdrawn.

Which one of the following describes the current control rod insertion capability?

A. **Manual insertion capability is NOT available using RMCS. Scram function is fully assured.**

B. **Manual insertion capability is available using RMCS. Scram function may be degraded.**

C. **Manual insertion capability is NOT available using RMCS. Scram function may be degraded.**

D. **Manual insertion capability is available using RMCS. Scram function is fully assured.**

Distracter Analysis:

A. **Incorrect**: The first half is correct since RMCS is not available; however scram function is not fully assured. See explanation for answer C.

B. **Incorrect**: The first half is incorrect since RMCS is not available without drive pressure. The second half is correct since scram function may be degraded. See explanation for answer C.

C. **Correct:** With both CRD pumps off, there is no drive pressure, so manual insertion using RMCS is not possible. Furthermore, CRD charging water pressure is also lost. At least 5 withdrawn control rods have low accumulator pressures rendering them inoperable. In addition, reactor pressure is less than 900 psig, so reactor pressure alone may not complete the scram. Rods may or may not fully insert or they may not insert fast enough under the given conditions, so the scram function may be degraded.

D. **Incorrect**: Both parts of this answer are wrong. See the explanation for answer C.

Level: RO Exam

Lesson Plan Objective: CRMECHE002
(R) Given a simplified diagram of a CRDM, explain the flowpath of water through the mechanism during the following modes of operation:
a. Insert
b. Withdrawal
c. Scram
  1) With Accumulator Water
  2) With Reactor Water
  d. No Rod Motion (Cooling)

Source: Modified INPO Bank (Cooper Question)

Level of knowledge: Analysis

Reference(s):
HC.OP-AB.IC-0001, "Control Rod"
Tech Spec LCO 3.1.3.5 and Bases

KA: 295022.K1.01
Knowledge of the operational implications of the following concepts as they apply to LOSS OF CRD PUMPS: (CFR: 41.8 to 41.10)
AK1.01 Reactor pressure vs. rod insertion capability. 3.3 3.4

Question: 26 Answer: B

1 Pt(s) A Design Basis LOCA has occurred and the plant responds in accordance with Design Basis assumptions.

Which of the following describes the response of the Reactor Building Area Radiation Monitors (ARMs) and the reason for those indications?

The ARMs......

A. show a slight increase in radiation levels due to the effective shielding provided by primary containment.

B. are upscale due to detector saturation as a result of the expected shine from the high radiation levels inside primary containment.

C. show a slight increase in radiation levels due to the slightly higher than normal radiation levels inside primary containment.

D. are upscale due to detector saturation as a result of the expected high radiation levels from fuel melting and primary to secondary containment leakage.

Distracter Analysis:

A. **Incorrect**: During post accident scenarios, all Reactor Building ARMs will be upscale and will remain so for about 30 days after the event.
Plausible: The containment does provide shielding.

B. **Correct:** During post accident scenarios, all Reactor Building ARMs will be upscale and will remain so for about 30 days after the event.

C. **Incorrect**: During post accident scenarios, all Reactor Building ARMs will be upscale and will remain so for about 30 days after the event. Radiation levels in containment are expected to be significantly higher than normal after a DBA LOCA.
Plausible:

D. **Incorrect**: Fuel melting is NOT expected during a DBA LOCA.
Plausible: The detectors will be upscale and it is plausible that someone could have the misconception that a DBA LOCA will result in fuel melting.

Level: RO Exam

Lesson Plan Objective: RADMONE002
From memory, briefly explain how plant radiation levels can be used to determine the extent of core damage.

Source: HC Bank (Audit Exam 1999)
Note: Slightly modified answer sentence structure and wording to improve plausibility. This is still considered a bank question.

Level of knowledge: Memory

Reference(s):
Lesson Plan NOH01RADMON, "Process and Area Radiation Monitoring Response"

KA: 295033.K2.02
Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS and the following: (CFR: 41.7 / 45.8) EK2.02 Process radiation monitoring system. 3.8 4.1

Comments / Change Record:

- Reworded answers to resolve plausibility concerns. (Chief & NRC reviewer comments 07/16)
- Deleted reference to "ALL" Reactor Building Radiation Monitors in the stem. (licensee comment 07/16)
- Added "and the plant responds in accordance with Design Basis assumptions" to the initial condition provided in the stem. (licensee comment 09/11)

Question: 27 Answer: B

1 Pt(s) The Unit is operating at 100% power with RCIC out of service for maintenance. A large seismic event occurs resulting in the following:

- A loss of offsite power has occurred and all EDG's are powering their respective buses.
- All control rods are fully inserted.
- HPCI is running and controlling RPV level at + 25 inches.
- Suppression pool level is 69 inches and lowering due to a leak on the HPCI pump suction line.
- RWCU Pipe Chase temperature is 165°F and rising due to a leak which CANNOT be isolated.
- The HPCI room water level is 5 inches and rising.
- The RCIC room water level is 5 inches and rising. .
- RPV pressure is 900 psig and stable.
- An Emergency Depressurization is ordered.

Which of the following describes the reason for the Emergency Depressurization?

A. **HPCI must be secured with room water level above the Max Safe Operating water level; therefore, low pressure ECCS systems will be required for level control.**

B. **A primary reactor coolant system is discharging into the reactor building with the HPCI & RCIC rooms above the Max Safe Operating water level.**

C. **The primary containment system is discharging into the reactor building with the HPCI & RCIC rooms above the Max Safe Operating water level.**

D. **The primary containment & a primary reactor coolant system are both discharging into the reactor building with the RWCU Pipe Chase above the Max Safe Operating temperature.**

Distracter Analysis:

A. **Incorrect**: Even if HPCI did need to be secured, then an emergency depressurization would not be required at this point since RPV level is at + 25 inches, and SLC is still available for high pressure injection. Even if there were a total loss of all high pressure injection sources, then an emergency depressurization would not be required until level reached – 185 inches.

B. **Correct**: EOP-103 requires an emergency depressurization when a primary coolant system is discharging into the reactor building and more than one area exceeds the Max Safe Operating level for the same parameter. In this case RWCU (part of the RCS) is actively discharging into the reactor building. In addition, The Max Safe Operating level has been exceeded in two different areas (HPCI room & RCIC room).

C. **Incorrect**: The leak in primary containment system leak is the cause of the high water level in the HPCI & RCIC rooms; however, the potential primary containment problem is not a direct reason for the emergency depressurization in EOP-103. The combination of high water level in the HPCI & RCIC rooms combined with the RWCU leak (primary system discharging into the reactor building) is the reason for the emergency depressurization.

D. **Incorrect**: One area above Max Safe Operating temperature is not a sufficient reason to perform an emergency depressurization. See justification for answer B for the requirements for an emergency depressurization.

Level: RO Exam

Lesson Plan Objective: EOP103E006
(R) Given any step in the procedure, describe the reason for performance of that step and/or expected system response to control manipulations prescribed by the step.

Source: New

Level of knowledge: Memory

Reference(s):
EOP-103 & Bases

KA: 295036.K3.01
Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT HIGH SUMP/AREA WATER LEVEL : (CFR: 41.5 / 45.6) EK3.01 Emergency depressurization.
2.6 2.8

Comments / Change Record:

- Added values for RWCU Pipe Chase Temperature and HPCI & RCIC room water levels rather than state that they are above max safe. (Chief comment 07/16)

Question: 28 Answer: A

1 Pt(s) Given the following:

- A large break LOCA occurred 4 minutes ago.
- RPV pressure is 125 psig and dropping.
- All ECCS is injecting.
- RPV level is - 120 inches and rising slowly.
- The "B" RHR pump is exhibiting symptoms of suction strainer clogging and may have inadequate NPSH.

Which of the following describes the correct action to take at this time?

A. Throttle HV-F048B, "RHR HX B Shell Side Byp MOV" AND HV-F003B, "B RHR HX Shell Side Outlet MOV" to reduce "B" LPCI injection flow.

B. Throttle HV-F017B, "RHR LOOP B LPCI INJ MOV" to reduce "B" LPCI injection flow.

C. Shift "B" RHR suction from the suppression pool to the CST.

D. Maximize all ECCS injection flow until RPV level is restored above + 12.5 inches.

Distracter Analysis:

A. **Correct:** HC.OP-AB-ZZ-0155 provides an option to throttle flow using HV-F003B & HV-F048 as needed to maintain NPSH. This action is permitted as long as RPV level can be maintained above TAF. In this case, level is above TAF and trending up indicating that there is margin to reduce injection flow.

B. **Incorrect**: HV-F017 is NOT a throttle valve and it is not discussed in HC.OP-AB-ZZ-0155 as an option to reduce RHR flow.

C. **Incorrect**: HC.OP-AB-ZZ-0155 provides this as an option for "B" Core Spray, but there is no CST suction path for RHR.

D. **Incorrect**: If RPV level cannot be maintained above TAF, then ECCS injection flow must be maximized regardless of NPSH; however, in the given scenario, RPV level is well above TAF and is rising indicating that there is margin to reduce flow on "A" LPCI without jeopardizing adequate core cooling.

Level: RO Exam

Lesson Plan Objective: RHRSYSE018
(R) Given plant conditions involving a Degraded ECCS Performance/Loss of NPSH, summarize required actions to mitigate the condition IAW HC.OP-AB.ZZ-0155 "Degraded ECCS Performance/Loss of NPSH."

Source: New

Level of knowledge: Analysis

Reference(s):
HC.OP-AB-ZZ-0155, "Degraded ECCS Performance / Loss of NPSH"

KA: 203000.A2.01
Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC); and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.01 Inadequate net positive suction head 3.2 3.4

Comments / Change Record:

- Changed initial condition to specify that the LOCA occurred 4 minutes ago.
- Changed answer "A" to throttle F003B & F048B.
- Changed distracter "B" to throttle F017B rather than F003B.
- Above changes made since the former rev had no correct answer as identified by licensee comment 07/16.
- Changed the RPV level given in the stem from -145 inches to -120 inches. (licensee comment 09/11)
- Reclassified cognitive level from memory to analysis. The question requires the ability to assemble multiple pieces of information to determine the correct course of action to take to solve the NPSH problem as well as apply system knowledge to pick the correct system manipulation to actually correct the problem. (NRC Exam Author 09/14)

Question: 29 Answer: B

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1 Pt(s) RHR Loop B is operating in the Shutdown Cooling (SDC) Mode, when RPV pressure rises to 83 psig.

Which one of the following describes the expected system response? (Assume all systems being operated from the control room.)

A. **RHR pump suction valve (F006B) auto closes, RHR Pump 'B' trips, and shutdown cooling return isolation valve (F015B) auto closes**

B. **Shutdown cooling suction valves (F008 and F009) auto close, RHR Pump 'B' trips, and shutdown cooling return isolation valve (F015B) auto closes**

C. **Shutdown cooling suction valves (F008 and F009) auto close, RHR pump suction valve (F006B) auto closes, and RHR Pump 'B' trips**

D. **RHR Pump 'B' trips, min flow control valve (F007B) auto opens, and shutdown cooling return isolation valve (F015B) auto closes.**

Distracter Analysis:

A. **Incorrect:** There is no auto closure interlock for F006B, at 82# (rising): F008/F009 close; Pump trips; F015 closes.
B. **Correct:** A SDC isolation has occurred at >82 psig.
C. **Incorrect**: There is no auto closure interlock for F006B, at 82# (rising): F008/F009 close; Pump trips; F015 closes.
D. **Incorrect**: F007 enabled only if pump breaker is closed, at 82# (rising): F008/F009 close; Pump trips; F015 closes

Level: RO Exam

Lesson Plan Objective: RHRSYSE011

Source: HC Bank (Q56406)

Level of knowledge: Memory

Reference(s):
HC.OP-SO.BC-0002

KA: 205000.A3.01
Ability to monitor automatic operations of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) including:
(CFR: 41.7 / 45.7) A3.01 Valve operation 3.2 3.1

Comments / Change Record:

- None

Question: 30 Answer: C

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1 Pt(s) The Unit was operating at 100% power when a Group I isolation occurred. HPCI status is as follows:

- HPCI is operating in the pressure control mode.
- The HPCI flow controller FIC-R600 is in AUTOMATIC and controlling flow at 3000 GPM.
- The HPCI TURBINE TROUBLE alarm was just received due to turbine oil cooler high temperature.
- HPCI turbine bearing temperatures are reading 155°F and rising slowly on Control Room recorder TR-R605.
- The Reactor Operator identifies that HPCI speed is 2000 RPM.

Which one of the following actions is required to be taken to attempt to correct the high temperature conditions so that HPCI can remain in service?

A. **Throttle HV-F008, "TEST BYP TO CST MOV" in the OPEN direction to RAISE turbine speed.**

B. **Throttle HV-F008 "TEST BYP TO CST MOV" in the OPEN direction to LOWER turbine speed.**

C. **Throttle HV-F008 "TEST BYP TO CST MOV" in the CLOSED direction to RAISE turbine speed.**

D. **Throttle HV-F008 "TEST BYP TO CST MOV" in the CLOSED direction to LOWER turbine speed.**

Distracter Analysis:

A. **Incorrect**: Throttling the valve open will lower flow resistance; therefore, turbine speed will lower as the FIC attempts to maintain the 3000 gpm set point. Lowering speed is a non-conservative action which will make the temperature problem worse and could lead to turbine bearing failure. See explanation for answer C.

B. **Incorrect**: Opening the HV-F008 valve will lower speed, but the correct action is to raise turbine speed above 2150 RPM. See explanation for answer C.

C. **Correct:** HC.OP-SO.BJ-0001, "High Pressure Coolant Injection System Operation" precaution 3.1.1 warns the operator to minimize HPCI turbine operations less when turbine speed is less to 2150 RPM in order to prevent turbine bearing or exhaust check valve chatter, so turbine speed needs to be raised in this scenario. Throttling the CST return valve in the closed direction increases the resistance to flow. The FIC is set to automatically control flow at 3000 gpm; therefore, the controller will call for an increase in turbine speed to maintain 3000 gpm. The increase in turbine speed will also increase the speed of the turbine driven oil pump which should resolve the bearing high temperature and oil cooler temperature. So, answer C is correct.

D. **Incorrect**: Closing HV-F008 will cause turbine speed to rise not lower. See explanation for answer C.

Level: RO Exam

Lesson Plan Objective: HPCI00E007
From memory, state the bases for HPCI turbine speed limitations during operation IAW HC.OP-SO.BJ-0001

Source: New

Level of knowledge: Analysis

Reference(s):

- HC.OP-SO.BJ-0001, "High Pressure Coolant Injection System Operation"
- HC.OP-AB.ZZ-0001, "Transient Plant Conditions"
- NOH01HPCI00, "High Pressure Coolant Injection System"

KA: 206000.A4.03
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.03 Turbine temperatures: BWR-2,3,4 3.1 3.0

Comments / Change Record:

- Changed the word "should" to "is required to" in the stem question. (licensee comment 07/16)

Question: 31 Answer: D

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1 Pt(s) Given the following:

- The Unit is in OPCON 1.
- ECCS Jockey Pump 1DP228 tripped.
- The CORE SPRAY LOOP B TROUBLE alarm was received due to low injection line pressure.

Which system can be lined up to clear the alarm condition, and where does this system inject into the Core Spray loop?

A. DEMIN WATER, which injects DOWNSTREAM of HV-F005B, "Core Spray Loop Inboard Isolation Valve".

B. DEMIN WATER, which injects UPSTREAM of HV-F005B, "Core Spray Loop Inboard Isolation Valve".

C. CONDENSATE TRANSFER, which injects DOWNSTREAM of HV-F005B, "Core Spray Loop Inboard Isolation Valve".

D. CONDENSATE TRANSFER, which injects UPSTREAM of HV-F005B, "Core Spray Loop Inboard Isolation Valve".

Distracter Analysis:

A. **Incorrect**: Demin water is not correct. Piping is installed from condensate transfer to each Core Spray loop, and a normally closed manual valve can be opened to provide keep fill to Core Spray from condensate transfer.

B. **Incorrect**: Demin water is not correct. Piping is installed from condensate transfer to each Core Spray loop, and a normally closed manual valve can be opened to provide keep fill to Core Spray from condensate transfer.

C. **Incorrect**: This is the correct system; however the injection point is upstream of HV-F005B.

D. **Correct:** See P&ID M-52-1.

Level: RO Exam

Lesson Plan Objective: CSSYS0E005

Source: New

Level of knowledge: Memory

Reference(s):
M-52-1, "Core Spray P&ID"
M-51-1, "RHR P&ID"
HC.OP-AR.ZZ-0007, "Overhead Annunciator Window Box B3"

KA: 209001.K1.03
Knowledge of the physical connections and/or cause effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
K1.03 Keep fill system 2.9 3.0

Comments / Change Record:

- Changed "should" to "can" in the stem question. (licensee comment 07/16)

Question: 32 Answer: B

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1 Pt(s) The plant is operating at 100% power when CORE SPRAY LINE BREAK (B3-B5) annunciates. This alarm could be the result of:

A. An inadvertent trip of ECCS jockey pump CP228 (RHR A/C and CSS A)

B. An inadvertent SLC injection

C. Placing the HPCI system in full flow test

D. Placing Core Spray Loop A in full flow test while Core Spray loop B is idle

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Distracter Analysis:

A. **Incorrect**: Sensors detect differential pressure downstream of testable check valves HV-F006A/B. With a loss of keep fill "A" loop, the upstream side of HV-F006A/B would see lower pressure but there is no change to downstream conditions.

B. **Correct:** The alarm response procedure lists this as a cause. SLC injects into the CS loop and the injection flow is sufficient to generate a DP which will activate the sensor that brings in the alarm.

C. **Incorrect**: HPCI injection would cause this alarm but not full flow test mode.

D. **Incorrect**: Sensors detect differential pressure downstream of testable check valves HV-F006A/B. Placing a loop in full flow test would not effect the pressure sensed downstream of HV-F006A/B.

Level: RO Exam

Lesson Plan Objective: CSSYS0E010
From memory, summarize/identify the purpose of the Core Spray System line break detection system, IAW the Core Spray System Lesson Plan.

Source: Modified HC Bank (Q56570)
Added a new answer (SLC injection) and this answer is the new correct answer.

Level of knowledge: Comprehension

Reference(s):

HC.OP-AR.ZZ-0007 for alarm window B3-B5

KA: 211000.K3.02
Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on following: (CFR: 41.7 / 45.4) K3.02 Core spray line break detection system: Plant-Specific 3.0* 3.2*

Comments / Change Record:
- None

Question: 33 Answer: D

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1 Pt(s) Given the following conditions:

- Reactor is critical.
- RPS bus A is being powered from the backup power supply.
- While restoring RPS bus A to the normal supply, the RPS Bus Transfer Switch is taken from "ALT A" through "NORMAL" to "ALT B".

SELECT the plant response:

A. 1/2 scram on RPS B, RPS A remains energized.

B. 1/2 scram on RPS A, RPS B remains energized.

C. RPS A and RPS B remain energized.

D. Full reactor scram.

Distracter Analysis:

A. **Incorrect**: RPS "A" would deenergize when placed from ALT A to NORMAL.
B. **Incorrect**: RPS "B" would deenergize when placed from NORMAL to ALT B.
C. **Incorrect**: Both channels would be deenergized.
D. **Correct:** Transferring from ALT A to NORMAL will cause a 1/2 scram on channel "A" then going to ALT B will cause a 1/2 scram on channel "B". The switch is break before make.

Level: RO Exam

Lesson Plan Objective: RPS000E006
Given the appropriate system operating procedure explain the effects on the reactor protection system when the power source is transferred from normal to alternate, and vice versa, IAW the RPS System Operating Procedure.

Source: HC Bank (Q54734)

Level of knowledge: Comprehension

Reference(s): HC.OP-AB.IC-0003

KA: 212000.K4.04
Knowledge of REACTOR PROTECTION SYSTEM design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) K4.04 The prevention of supplying both RPS buses simultaneously from the alternate power source: Plant-Specific 3.1 3.1

Comments / Change Record:

- Deleted words "MCC 10B491 through transformer 1AX432". (licensee comment)
- Changed the cognitive level classification from memory to comprehension. The question tests knowledge of the consequences of operating the equipment in the manner presented in the question. (NRC Exam Author 09/14)

Question: 34 Answer: D

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1 Pt(s) Complete the following statements:

The IRM detectors operate in the ________ region of the amplification curve. In addition, a decrease in IRM detector argon gas pressure will cause the IRM detectors to be _________ sensitive.

A. proportional / more

B. proportional / less

C. ionization / more

D. ionization / less

Distracter Analysis:

A. **Incorrect**: Proportional region is incorrect. In addition, argon is used as a detector ionization gas, so reduced argon gas pressure will yield less ionization events; therefore, the detector is less sensitive.
Plausible: The SRMs operate in the proportional region. In addition, argon could be confused for a quench gas, and if it did act as a quench gas, then the detector would be more sensitive with less quench gas.

B. **Incorrect**: Proportional region is incorrect.
Plausible: The SRMs operate in the proportional region. In addition, the second half of the answer is correct.

C. **Incorrect**: Argon is used as a detector ionization gas, so reduced argon gas pressure will yield less ionization events; therefore, the detector is less sensitive.
Plausible: Ionization region is correct. In addition, argon could be confused for a quench gas, and if it did act as a quench gas, then the detector would be more sensitive with less quench gas.

D. **Correct:** IRM detectors operate in the ionization region. Argon is used as a detector ionization gas, so reduced argon gas pressure will yield less ionization events; therefore, the detector is less sensitive.

Level: RO Exam

Lesson Plan Objective: IRMSYSE003
Given a drawing of an IRM Detector, label the simplified drawing of the fission chamber and explain the principles of operation of the detector, IAW Control Room Procedures.

Source: New

Level of knowledge: Comprehension

Reference(s):
Lesson Plan NOH01IRMSYS, "IRM System"

KA: 215003.K5.01
Knowledge of the operational implications of the following concepts as they apply to INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : (CFR: 41.5 / 45.3) K5.01 Detector operation 2.6 2.7

Comments / Change Record:

- Reclassified cognitive level from memory to comprehension. The detector must be recalled solely from memory; however, understanding how detector sensitivity changes as a function of gas pressure requires a higher level comprehension of detector operation. (NRC Exam Author 09/14)

Question: 35 Answer: C

1 Pt(s) Given the following:

- The Unit is in OPCON 2.
- The reactor is critical.
- SRM shorting links are installed.
- Reactor power is on Range 3 of the IRMs.
- The "A" Source Range Monitor (SRM) is being withdrawn from the core.
- During the SRM withdrawal, ±24 VDC SRM power is lost on the "A" SRM with 0 VDC indicated at 1AD307.

Which one of the following describes the SRM system response?

A. An SRM rod block is generated; "A" SRM detector stops withdrawing from the core.

B. An SRM rod block is NOT generated; "A" SRM detector stops withdrawing from the core.

C. An SRM rod block is generated; "A" SRM detector continues to withdraw from the core.

D. An SRM rod block is NOT generated; "A" SRM detector continues to withdraw from the core.

Distracter Analysis:

A. **Incorrect**: The first half of the answer is correct since the SRM inop trip is active up to IRM Range 8 with the mode switch in startup, so a rod block is generated. However, drive power is from 10Y202 AC power not DC power, so the SRM's continue to withdraw.
Plausible: The first half of the answer is correct. The second half of the answer is a plausible incorrect answer and knowledge of the power supplies is needed to select the right answer.

B. **Incorrect**: 24 VDC supplies the high voltage power supply, so a loss of 24 VDC results in an inop trip. The SRM inop trip is active up to IRM Range 8 with the mode switch in startup, so a rod block is generated. Furthermore, drive power is from 10Y202 AC power not DC power, so the SRM's continue to withdraw.
Plausible: The first half of the answer is plausible and knowledge of the logic and interlocks is required. The second half of the answer is

a plausible incorrect answer and knowledge of the power supplies is needed to select the right answer.

C. **Correct:** 24 VDC supplies the high voltage power supply, so a loss of 24 VDC results in an inop trip. The SRM inop trip is active up to IRM Range 8 with the mode switch in startup, so a rod block is generated. Drive power is from 10Y202 AC power not DC power, so the SRM's continue to withdraw.

D. **Incorrect**: 24 VDC supplies the high voltage power supply, so a loss of 24 VDC results in an inop trip. The SRM inop trip is active up to IRM Range 8 with the mode switch in startup, so a rod block is generated.
Plausible: The first half of the answer is a plausible incorrect answer and knowledge of the logic and interlocks is required. The second half of the answer is correct.

Level: RO Exam

Lesson Plan Objective: SRMSYSE013
From memory, predict the SRMS response to loss of operating supply voltages, IAW the Student Handout.

Source: INPO Bank (Perry 1)

Level of knowledge: Analysis

Reference(s):

- HC-OP.AB.ZZ-0151, "24 Volt DC Malfunction"
- Lesson Plan NOH01DCELEC, "DC Electrical Distribution"
- Lesson Plan NOH04SRMSYS, "SRM System"

KA: 215004.K6.02
Knowledge of the effect that a loss or malfunction of the following will have on the SOURCE RANGE MONITOR (SRM) SYSTEM : (CFR: 41.7 / 45.7) K6.02 24/48 volt D.C. power 3.1 3.3

Comments / Change Record:

- Added initial condition that shorting links are installed. (licensee comment 07/16)
- Revised question to reflect that ONLY the "A" SRM is being withdrawn based on plant practice to withdraw one SRM at a time. Also, removed reference to "C" SRM in all of the answers. (licensee comment 09/11)
- Reclassified cognitive level from memory to analysis. The question tests ability to predict the effect of a loss of power

on both the SRM rod block logic as well as drive control power.

Question: 36 Answer: D

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1 Pt(s) Given the following:

- A reactor scram signal has occurred
- The full core display has several full-in lights NOT displayed
- CRIDS/SPDS/RWM computer rod position data is NOT available

You are directed to verify the scram and you are aware of industry experiences where several light bulbs on the full core display have failed concurrently.

Which of the following indications would cause you to conclude that the scram may NOT have been successful?

A. All LPRM "downscale" lights are illuminated.

B. Reactor period stable at –80 seconds on SRMs.

C. RPS channel trip logic lights are extinguished.

D. All APRM "downscale" lights are extinguished.

Distracter Analysis:

A. **Incorrect**: LPRM downscale lights are not a conclusive indication that the reactor is shutdown; however, all lights lit should are a fairly strong indicator that the reactor is shutdown.
Plausible:

B. **Incorrect**: -80 second period is an indication for a shutdown reactor.
Plausible:

C. **Incorrect**: RPS trip lights indicate that RPS is tripped; however, it does not provide any assurance that the rods actually went in.
Plausible:

D. **Correct**: All APRM downscale lights extinguished means that power is greater than 4% power; therefore, the reactor is NOT shutdown.

Level: RO Exam

Lesson Plan Objective: AB0000E003

Source: Hope Creek Bank
Last NRC Exam 1999

Level of knowledge: Comprehension

Reference(s):

KA: 215005.A1.02
Ability to predict and/or monitor changes in parameters associated with operating the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM controls including: (CFR: 41.5 / 45.5) A1.02 RPS status 3.9 4.0

Comments / Change Record:

- None

Question: 37 Answer: B

1 Pt(s) Given the following:

- The Unit was operating at 100% power.
- A total Station Blackout is in progress.
- HPCI failed and is unavailable.
- RCIC is injecting at low flow with RPV level at + 35 inches.
- RPV pressure is 120 psig and dropping.
- HC.OP-AB.ZZ-0135 and the appropriate EOPs are being executed.

Complete the following statement:

RCIC will automatically isolate when RPV steam supply pressure reaches approximately _________psig, AND the Emergency Operating Procedures ____________ the isolation to be defeated.

A. 65 ; do NOT allow

B. 65 ; allow

C. 100 ; do NOT allow

D. 100 ; allow

Distracter Analysis:

A. **Incorrect**: EOP-0101 lists RCIC as a preferred injection source and EOP-0101 Table 1 states, "If necessary, bypass the low pressure isolation".
Plausible: Setpoint is correct AND the EOPs do NOT allow the low steam supply pressure isolation to be defeated for HPCI, so it is plausible that someone could conclude the same for RCIC.

B. **Correct:** The nominal RCIC low steam supply pressure isolation setpoint is approximately 65 psig. EOP-0101 lists RCIC as a preferred injection source and EOP-0101 Table 1 states, "If necessary, bypass the low pressure isolation".

C. **Incorrect**: Setpoint is incorrect AND EOP-0101 lists RCIC as a preferred injection source and EOP-0101 Table 1 states, "If necessary, bypass the low pressure isolation".
Plausible: 100 psig is the HPCI low pressure isolation setpoint AND the EOPs do NOT allow the low steam supply pressure

isolation to be defeated for HPCI, so it is plausible that someone could conclude the same for RCIC.

D. **Incorrect**: Setpoint is incorrect.
Plausible: 100 psig is the HPCI low pressure isolation setpoint. AND the RCIC isolation may be defeated.

Level: RO Exam

Lesson Plan Objective: RCIC00E009
(R) Given plant conditions, determine the signal(s) which will cause the RCIC System to automatically isolate, IAW the system operating procedure.

Source: New

Level of knowledge: Memory

Reference(s):

- Lesson Plan NOH04RCIC00, "RCIC System"
- HC.OP-AB.ZZ-0135
- HC.OP-SO.BD.0001
- EOP-0101

KA: 217000.A2.03
Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.03 Valve closures 3.4 3.3

Comments / Change Record:

- Total rewrite due to licensee comments on the min flow valve question. (licensee comment 07/16)

Question: 38 Answer: A

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1 Pt(s) The following plant conditions exist at T = 0:

- Reactor water level is –129 inches and slowly dropping
- Reactor pressure is 900 psig
- Drywell pressure is 1.40 psig and steady
- "A" RHR pump is running
- All of the MSIVs are closed
- ADS has NOT been inhibited

Which of the following describes how ADS will respond under these conditions?

A. **ADS will initiate at T = 405 seconds.**

B. **ADS will initiate at T = 300 seconds.**

C. **ADS will initiate at T = 105 seconds.**

D. **ADS will NOT initiate.**

Distracter Analysis:

A. **Correct:** The high drywell pressure bypass timer is activated at -129 inches, so ADS will initiate after the 5 minute (300 second) high drywell pressure bypass timer times out PLUS the 105 second ADS timer. 405 total seconds until initiation. One RHR pump running is sufficient to support ADS initiation.

B. **Incorrect:** ADS initiates in 405 seconds in this scenario. See justification for answer A.
Plausible: 300 seconds is the time for the high drywell pressure bypass timer. This timer is activated in this scenario, but the 105 second ADS timer must also time out after the 300 second bypass timer times out. It is possible that someone could conclude that 300 seconds is the time to initiation.

C. **Incorrect**: ADS initiates in 405 seconds in this scenario. See justification for answer A.
Plausible: -129 inches alone will initiate ADS but only after the 300 second high drywell pressure bypass timer times out followed by the 105 second ADS timer. In addition, if high drywell pressure coincidence were met (1.68 psig), then ADS would initiate in 105 seconds.

D. **Incorrect**: ADS initiates in 405 seconds in this scenario. See justification for answer A.
Plausible: A candidate could believe that -129 inches and 1.68 psig are both required for ADS to initiate. In addition, a sufficient number of low pressure ECCS pumps need to be running for ADS to initiate (1 RHR or 2 CS pumps), so someone could think that just 1 RHR pump is not enough.

Level: RO Exam

Lesson Plan Objective: ADSSYSE004
From memory, list/identify the five (5) signals (including set points) which will cause the Automatic Depressurization System to automatically initiate, IAW the Automatic Depressurization System Lesson Plan.

Source: HC Bank (Q56872)

- NRC exam 12/98
- Slightly modified. Stem was changed to have only 1 RHR pump running rather than all ECCS. Distracter B was changed from "ADS initiates at time 0" to "ADS initiates at T = 300 seconds".

Level of knowledge: Memory

Reference(s):
HC.OP-SO.SN-0001

KA: 218000.A3.04
Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: (CFR: 41.7 / 45.7)
A3.04 Primary containment pressure 3.7 3.8

Comments:

- None

Question: 39 Answer: D

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1 Pt(s) Given the attached SPDS display screen shot:

What is the plant status based on the provided SPDS indication?

A Level 2 isolation signal.......

A. **has NOT been received, AND SPDS has lost position indication monitoring capability on at least one valve.**

B. **has NOT been received, AND at least one monitored valve is open.**

C. **has been received, AND ALL monitored valves isolated as required.**

D. **has been received, AND at least one monitored valve failed to isolate as required.**

Distracter Analysis:

A. **Incorrect**: See correct answer.
Plausible: NISOL could be interpreted as NO isolation. Red indication could be interpreted as a "a problem" and loss of monitoring capability is a plausible problem.

B. **Incorrect**: See correct answer.
Plausible: NISOL could be interpreted as NO isolation. The indication provided does mean that at least one monitored valve failed to isolate (it is open).

C. **Incorrect**: At least one valve failed to isolate as required.
Plausible: Isolation signal has been received. It is plausible that someone could misinterpret the indication to mean that all valves are isolated. For instance, NISOL = "normal isolation" is one possible interpretation.

D. **Correct:** NISOL in red indication means that an isolation signal has been received and that at least one valve has failed to isolate as required.

Level: RO Exam

Lesson Plan Objective:

- NSSSSOE006: Given a copy of the SPDS containment isolation status, evaluate the status of the NSSSS generated isolation signals IAW the NSSSS student handout.
- NSSSSOE014: Given a specific parameter, which initiates NSSSS isolation signals, identify all valves isolated by that parameter and the set point at which that isolation signal is generated.

Source: New

Level of knowledge: Memory

Reference(s):

- Lesson Plan NOH04SPDS00C, "SPDS"
- Lesson Plan NOH04NSSSS0C, Nuclear Steam Supply Shutoff System"

KA: 223002.A4.05
Ability to manually operate and/or monitor in the control room: (CFR: 41.7 / 45.5 to 45.8) A4.05 SPDS/ERIS/CRIDS/GDS: Plant-Specific . 2.5* 2.8*

Comments:

- Rewrote question based on Chief Examiner and licensee comments to provide an SPDS screen shot and then have the candidate determine what the screen shot means. (Chief & licensee 07/16)

Question: 40 Answer: C

1 Pt(s) Given the following conditions:

- Power ascension is in progress following a refuel outage
- Reactor power is 97%
- PSV-F013P opens inadvertently and does **NOT** reclose

Select the IMMEDIATE operator action:

A. Depress the "Reset Logic Armed" pushbutton for "B" Low-Low set logic.

B. Lock the mode switch to SHUTDOWN.

C. Reduce reactor power to 95%.

D. Dispatch the operator to remove the SRV fuses.

Distracter Analysis:

A. Incorrect: Subsequent action.
B. Incorrect: Retainment override if unable to close the valve.
C. Correct: This is an immediate action per HC.OP-AB.RPV-0006.
D. Incorrect: Subsequent action.

Level: RO Exam

Lesson Plan Objective: ABRPV6E003
From memory, recall the Immediate Operator Actions for Safety/Relief Valve.

Source: HC Bank (Q77604)

Level of knowledge: Memory

Reference(s):
HC.OP-AB.RPV-0006

KA: 239002.G2.4.49
2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6)

Comments / Change Record:

- None

Question: 41 Answer: B

1 Pt(s) Given the following:

- The Unit is in OPCON 3 following a shutdown for refueling.
- RPV pressure is 900 psig.
- Feedwater level control is maintaining level at + 25 inches.
- An RPV cool down is in progress.

Complete the following statement:

The feed water level control system uses ________ range instrumentation, and indicated RPV level will read ________ than actual RPV level on that range as the RPV cools down.

A. **narrow / lower**

B. **narrow / higher**

C. **wide / lower**

D. **wide / higher**

Distracter Analysis:

A. **Incorrect**: Narrow range is correct; however, the second half is incorrect. See justification for answer B.

B. **Correct:** FWLCS uses the narrow range. If reactor vessel temperature is significantly below calibration conditions, variable leg density and pressure will be higher, causing indicated level to be higher than actual. The Integrated Operating Procedures contain attachments with Vessel Level Instrumentation Temperature Compensation Curves, which allow determination of actual level for off calibration conditions.

C. **Incorrect**: Wide range is incorrect and indicated level will be higher than actual. See justification for answer B.

D. **Incorrect**: Wide range is incorrect; however, the second part is correct. See justification for answer B.

Level: RO Exam

Lesson Plan Objective: RXINSTE007
Given any reactor vessel water level range (i.e., narrow, wide, etc.), determine each of the following:

- The calibration criteria (hot or cold)
- The operational conditions for which the range is intended
- Any Control Room instrumentation

Source: New

Level of knowledge: Comprehension

Reference(s):

- HC.OP-IO.ZZ-0004, "Shutdown from Rated Power to Cold Shutdown"
- Lesson Plan NOH04RXINSTC, "Nuclear Boiler Instrumentation"

KA: 259002.K1.09
Knowledge of the physical connections and/or cause effect relationships between REACTOR WATER LEVEL CONTROL SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8)
K1.09 P sat/T sat (compensation) .2.9 3.0

Comments / Change Record:

- Reclassified cognitive level from memory to comprehension. The first part of the question tests at the memory level; however, the second part requires comprehension of how density affects level indication. (NRC Exam Author 09/14)

Question: 42 Answer: D

1 Pt(s) Given the following:

- The Unit was operating at 100% power.
- RCIC is out of service for maintenance.
- A total station blackout event just occurred.
- RPV level dropped to – 65 inches.
- HPCI automatically injected to recover RPV level.
- RPV level is now stable at + 35 inches.

Which one of the following describes long term HPCI operations?

A. **HPCI will eventually trip on turbine high exhaust pressure as non condensable gases build up in the barometric condenser due to the loss of the barometric condenser vacuum pump discharge flow path. In addition, HPCI operation is undesirable without ventilation in service since airborne contamination levels may eventually rise in the Reactor Building and/or HPCI compartment.**

B. **HPCI will continue to operate since the barometric condenser vacuum pump discharge flow path will automatically swap to the main condenser due to high backpressure in the HPCI barometric condenser. The swap will preserve HPCI operation and prevent potential airborne contamination in the Reactor Building and/or HPCI compartment.**

C. **HPCI will continue to operate since the barometric condenser vacuum pump discharge flow path will automatically swap to the suppression chamber airspace due to high backpressure in the HPCI barometric condenser. The swap will preserve HPCI operation and prevent potential airborne contamination in the Reactor Building and/or HPCI compartment.**

D. **HPCI will continue to operate; however, airborne contamination levels may eventually rise in the Reactor Building and/or HPCI compartment as the barometric condenser vacuum pump continues to discharge into the RBVS/FRVS ductwork without ventilation in service.**

Distracter Analysis:

A. **Incorrect**: This condition will not cause high turbine exhaust pressure. The barometric condenser services HPCI turbine gland seal leak off.
Plausible: The candidate may incorrectly think that the barometric condenser is linked directly with the turbine main process flow. For instance, the main turbine discharges to the main condenser, and a loss of main condenser vacuum leads to a main turbine trip. In addition, the second statement in the answer is correct since it is not "desirable" to run HPCI without ventilation in service.

B. **Incorrect**: HPCI will continue to operate; however, there is no automatic swap feature to realign the vacuum pump discharge back to the main condenser.
Plausible: The HPCI vacuum pump normally discharges to the main condenser for test runs, and the discharge flow path automatically realigns to the RBVS/FRVS on a HPCI initiation signal.

C. **Incorrect**: HPCI will continue to operate; however, there is no automatic feature to swap the vacuum pump discharge to the suppression chamber air space.
Plausible: The HPCI turbine normally discharges to the suppression pool, and there are vacuum breakers on the discharge line that connect to the suppression chamber air space. In addition, it would almost make sense to discharge a potentially contaminated discharge path back to primary containment when secondary containment treatment systems are not available.

D. **Correct:** The DC powered HPCI barometric condenser vacuum pump will continue to discharge into the RBVS/FRVS ventilation duct. Without ventilation in service, potentially contaminated non condensable gases may eventually leak into the Reactor Building and/or HPCI compartment.

Level: RO Exam

Lesson Plan Objective:
HPCI00E006: From memory, summarize the interrelationship(s) between the HPCI System and any of the following IAW control room references (Condenser, RBVS, etc.).

Source: New

Level of knowledge: Comprehension

Reference(s):
Lesson Plan NOH01HPCI00, "High Pressure Coolant Injection"
P&ID M-55-1

KA: 261000.K3.04
Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: (CFR: 41.7 /45.6) K3.04 High pressure coolant injection system: Plant-Specific. 3.1 3.1

Note: Hope Creek uses FRVS (Filter Recirculation Ventilation System) rather than SBGTS. This is a plant specific feature at Hope Creek. Fundamentally, the FRVS & SBGTS systems perform a similar function which is to treat the secondary containment atmosphere and provide an elevated release point under accident conditions. The HPCI system interrelationship with FRVS is the same as for plants using SBGTS.

Comments / Change Record:

- Deleted statements in the stem indicating that RBVS and FRVS are out of service since this should be known from the blackout condition. (NRC reviewer comment 07/16)
- Deleted the word "best" from the stem question.

Question: 43 Answer: A

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1 Pt(s) Given the following:

- A station blackout has occurred
- All 500 KV Lines to Hope Creek are de-energized

IAW HC.OP-AB.ZZ-0135, Station Blackout/Loss of Offsite Power/Diesel Generator Malfunction, which one of the following 500 KV lines is requested first to be re-energized in order to restore power to Hope Creek 13 KV ring bus?

A. Salem 5037 Line

B. New Freedom 5024 Line

C. Red Lion 5015 Line

D. East Windsor 5021 Line

Distracter Analysis:

A. **Correct:** AB-135 power restoration strategy is to restore power via the Salem 5037 line and the Salem Gas Turbine.

B. **Incorrect**: See justification for A.

C. **Incorrect**: See justification for A.

D. **Incorrect**: See justification for A.

Level: RO Exam

Lesson Plan Objective: 0AB135E006
Explain the information contained in the Discussion Section of Station Blackout/Loss Of Offsite Power Diesel Generator Malfunction, Abnormal Operating Procedure.

Source: HC Bank (Q68928)
NRC Exam 2/2002

Level of knowledge: Memory

Reference(s): HC.OP-AB.ZZ-0135, 4.14

KA: 226001.K2.01
Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 Off-site sources of power 3.3 3.6

Comments / Change Record:

- Revised Deans Line to “East Windsor”. (licensee comment 08/27)

Question: 44 Answer: C

1 Pt(s) Given the following:

- 120 VAC uninterruptible power supply 1AD481 is aligned with the MANUAL BYPASS CONTROL SWITCH in "NORMAL".
- A loss of voltage is detected on the output of the inverter section.

Complete the following statement:

The static switch will automatically transfer to the _________ power supply, and the static switch ________ automatically swap back to the inverter once the inverter output voltage returns to normal.

A. alternate / will

B. alternate / will NOT

C. backup / will

D. backup / will NOT

Distracter Analysis:

A. **Incorrect**: The alternate source is an auctioneered supply on the input side of the inverter section. In this scenario, the inverter output section went dead, so both the normal and alternate supplies are not available. Therefore, the static switch will automatically swap to the backup source in this particular case. The second half is correct in that the static switch will swap to the "preferred source" (inverter output) once voltage returns to normal.

B. **Incorrect**: The alternate source is an auctioneered supply on the input side of the inverter section. In this scenario, the inverter output section went dead, so both the normal and alternate supplies are not available. Therefore, the static switch will automatically swap to the backup source in this particular case. The second half is also incorrect since the static switch will swap to the "preferred source" (inverter output) once voltage returns to normal.

C. **Correct:** In this scenario, the inverter output section went dead, so both the normal and alternate supplies are not available. Therefore, the static switch will automatically swap to the backup source in this particular case. In addition, the static switch is normal seeking and will swap to the "preferred source" (inverter output) once voltage returns to normal.

D. **Incorrect**: The first half is correct, but the second half is incorrect. See justification for answer C.

Level: RO Exam

Lesson Plan Objective: 1EAC00E020
Given a set of plant conditions, summarize/identify the UPS System response to the following:

- Loss of AC to the Rectifier Cabinet
- Loss of DC to the Rectifier Cabinet
- Loss of Inverter Output
- Loss of AC Regulator Output

Source: New

Level of knowledge: Analysis

Reference(s):
Lesson Plan NOH01EAC00, "Class 1E AC Power Distribution"

KA: 262001.K4.01
Knowledge of UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) K4.01 Transfer from preferred power to alternate power supplies 3.1 3.4

Comments / Change Record:

- Capitalized "NOT" in the answers. (licensee comment 07/16)
- Reclassified the cognitive level from memory to analysis since the question tests the ability to predict the outcome of a voltage transient on the inverter based on a specific equipment configuration. (NRC Exam Author 09/14)

Question: 45 Answer: A

1 Pt(s) Given the following:

- The Unit is in OPCON 1 at 100% power.
- The CONTROL AREA HVAC FAN MALF (E6-B1) is in alarm.
- A & BVH410, Control Area Battery Exhaust Fans have both failed and will be out of service for the next 24 hours.
- Hydrogen monitoring is being performed in the affected battery rooms once per shift in accordance with HC.OB-AB.HVAC-0001, "HVAC".
- During rounds, the responsible operator reports that the current HPCI battery room Hydrogen concentration is about 1%, and this H2 concentration is equivalent to 25% of the Lower Explosive Limit (LEL).

Which one of the following describes the correct course of action IAW HC.OB-AB.HVAC-0001?

A. **Initiate a fire impairment and prop open the doors to the HPCI battery room to reduce the Hydrogen concentration.**

B. **Secure the HPCI battery charger to stop H2 generation.**

C. **Place the HPCI battery charger in EQUALIZE mode to minimize H2 generation.**

D. **Initiate a fire impairment to maintain the HPCI battery room doors closed and to restrict access to the room until the exhaust fans are restored.**

Distracter Analysis:

A. **Correct:** This is the only option presented that both reduces the H2 concentration in the room while preserving battery and HPCI function. In addition, procedure HC.OB-AB.HVAC-0001, "HVAC" specifically directs that the doors be opened if H2 is above 10% of the LEL. The procedure also directs that a fire impairment be initiated due to the need to prop open the doors. .

B. **Incorrect**: This not a course of action directed by any procedure. This action would render the battery and HPCI inoperable (see Tech Spec 3.8.2.1 Action b.). It would also affect availability during any event requiring HPCI since the batteries would discharge.

Furthermore, the action does nothing to correct the existing high H2 concentration.
Plausible: Battery charging does result in H2 generation, so securing the charger may seem to be a possible solution.

C. **Incorrect**: Equalize mode places an overcharge condition on the battery; therefore, the H2 generation rate and concentration will increase.
Plausible: It is a mode of charger operation and could be selected if this mode of operation and its consequences were not understood.

D. **Incorrect**: Maintaining the current situation will allow the H2 concentration to increase to potentially flammable conditions.
Plausible: Containing the situation to the one room may seem like a good idea. Restricting access is a good idea in this situation.

Level: RO Exam

Lesson Plan Objective: DCELECE018
Given a set of plant conditions/malfunctions associated with battery ventilation, evaluate whether a loss of ventilation to a battery room can result in equipment failure. IAW Available Control Room References.

Source: New

Level of knowledge: Analysis

Reference(s):
- HC.OP-AB.HVAC-0001, "HVAC"
- Lesson Plan NOH01CAVENTC, "Control Area Ventilation"
- Lesson Plan NOH01DCELEC, "DC Electrical Distribution"

KA: 263000.K5.01
Knowledge of the operational implications of the following concepts as they apply to D.C. ELECTRICAL DISTRIBUTION: (CFR: 41.5 / 45.3) K5.01 Hydrogen generation during battery charging. .2.6 2.9

Comments / Change Record:
- Added IAW procedure in the stem question. (NRC reviewer comment 07/16)

Question: 46 Answer: D

1 Pt(s) Given the following:

- EDG-A is paralleled to 10A401 Bus for testing.
- The engine driven fuel oil pump experiences an internal failure such that the pump will NOT develop discharge pressure.

What will be the EDG response to this condition?

EDG output frequency will.....

A. remain constant UNTIL the diesel trips on reverse power.

B. lower AND THEN the diesel will trip on low fuel oil pressure.

C. lower AND THEN the diesel will trip on reverse power.

D. remain constant AND the diesel will continue to run.

Distracter Analysis:

A. **Incorrect**: The electric standby fuel oil pump will automatically start when fuel oil pressure drops to 20 psig, so the engine will continue to run as if nothing happened.
Plausible: If there were no standby fuel oil pump, then this would be the correct answer. Frequency would remain constant, the diesel would stop carrying load, and the generator would motor until it tripped on reverse power.

B. **Incorrect**: The electric standby fuel oil pump will automatically start when fuel oil pressure drops to 20 psig, so the engine will continue to run as if nothing happened.
Plausible: A plausible misconception is that frequency would lower as fuel starvation occurred. In addition, some engine designs have a low fuel oil pressure trip, but this one does not. It does have a low lube oil pressure trip. So, the second half of the answer is plausible but incorrect.

C. **Incorrect**: The electric standby fuel oil pump will automatically start when fuel oil pressure drops to 20 psig, so the engine will continue to run as if nothing happened.
Plausible: A plausible misconception is that frequency would lower as fuel starvation occurred. In addition, if there were no standby pump, then the diesel would in fact motor and trip on reverse power.

D. **Correct:** Frequency is locked in a grid frequency when the EDG is paralleled to the grid. More importantly, the electric standby fuel oil pump will automatically start when fuel oil pressure drops to 20 psig, so the engine will continue to run as if nothing happened.

Level: RO Exam

Lesson Plan Objective: EDG000E007
Given plant conditions, determine if the Diesel Generator will trip under manual and/or automatic start conditions.

Source: Mod HC Bank (Q25853)

- Stem condition was modified to change the malfunction from a governor problem to a fuel oil pump failure.
- Changed the answer to be that the engine would continue to run.
- Replaced one distracter. The new distracter states that the diesel will trip on low fuel oil pressure rather than generator overcurrent.

Level of knowledge: Analysis

Reference(s):
Lesson Plan NOH01EDG000C, "Emergency Diesel Generators"

KA: 264000.K6.02
Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET) : (CFR: 41.7 / 45.7) K6.02 Fuel oil pumps 3.6 3.6

Comments / Change Record:

- Eliminated "frequency and diesel" from the stem question. (licensee comment 07/16)
- Revised the second bullet in the stem to state "the engine driven fuel oil pump experiences an internal failure such that the pump will not develop discharge pressure" in place of "the engine driven fuel oil pump fails due to pump seizure". (licensee comment 09/11)
- Capitalized "NOT" in the last bullet in the stem. (licensee comment 09/13)
- Reclassified cognitive level from memory to analysis. The question tests the ability to predict diesel engine response given a loss of engine driven fuel pump while connected to the grid. (NRC Exam Author 09/14)

Question: 47 Answer: A

1 Pt(s) Given the following:

- The Unit is in OPCON 1.
- Instrument Air Dryer 00-F-104 is IN SERVICE.
- Instrument Air Dryer 10-F-104 is in STANDBY.
- HV-11416, "Instrument Air Dryer 1A-F-104 Isolation Valve" is CLOSED by placing hand switch HS-11416 in CLOSE.
- Instrument Air header pressure is at about 100 psig.

Then,

- A flow switch fails high causing an EXCESSIVE FLOW signal to be generated on the IN SERVICE Instrument Air Dryer 00-F-104.

Which one of the following describes the initial Instrument Air system response, and if required, any appropriate operator actions that may be needed to stabilize or restore Instrument Air header pressure?

A. **Instrument Air header pressure will slowly drop. Place HS-11416 in OPEN for HV-11416 "Instrument Air Dryer 1A-F-104 Isolation Valve".**

B. **Instrument Air header pressure will remain in the normal range since HV-11416 will automatically OPEN immediately upon receipt of the EXCESSIVE FLOW signal. NO operator action is required.**

C. **Instrument Air header pressure will slowly drop and HV-11416 will automatically OPEN when Instrument Air header pressure drops to 90 psig. NO operator action is required.**

D. **Instrument Air header pressure will slowly drop. JACK OPEN HV-11416 by isolating the vent off the bottom of the air piston since the valve will NOT open either automatically OR with the control switch as a result of the EXCESSIVE FLOW signal.**

Distracter Analysis:

A. **Correct:** The false EXCESSIVE FLOW signal will trip off the in service dryer (00-F-104) and will prevent the automatic swap to the standby filter (10-F-104). Air header pressure will slowly decay due to normal air usage. Since there is no leak, then the operators can place the third air dryer (1A-F-104) by opening the associated discharge valve (HV-11416) via the hand switch. This will stabilize and restore header pressure. Reference HC.OP.AB.COMP.

B. **Incorrect**: There is no automatic open feature based on EXCESSIVE FLOW.
Plausible: The correct action is to open HV-11416 under these circumstances, and there are some dryer malfunctions that will cause the valve to automatically open.

C. **Incorrect**: HV-11416 will not automatically open at 90 psig.
Plausible: Air header pressure will slowly drop and the valve will open automatically at 85 psig.

D. **Incorrect**: HV-11416 fails open on loss of air, so this action would actually force the valve to close rather than open.
Plausible: Instrument air header pressure will drop. The action would be correct to attempt to open the discharge valve for the dryer that was in service (00-F-104) or for the dryer that was in standby (10-F-104) as directed by HC.OP.AB.COMP.

Level: RO Exam

Lesson Plan Objective: INSAIRE013
From memory, describe the Instrument Air System dryer package response if the in-service dryer package were to receive an AUTO TRANSFER signal when any of the following conditions existed, available references.

- Standby dryer package has an alarm condition.
- Operating in-service dryer package has air excessive flow condition.
- Off-service dryer package control switch is not in the STANDBY position.

Source: New

Level of knowledge: Analysis

Reference(s):

- HC.OP-AB.COMP-0001, "Instrument Air and/or Service Air"
- NOH01INSAIR, "Instrument Air System"

KA: 300000.A2.01
Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: (CFR: 41.5 / 45.6) A2.01 Air dryer and filter malfunctions . 2.9 2.8

Comments / Change Record:

- Replaced "with" in the stem with "by placing". (NRC reviewer comment 07/16)
- Deleted word "expected" from the question stem and capitalized the word "NO" in the answers. (licensee comments 07/16)
- Reworded the last bullet in the stem to state "A flow switch fails high causing an EXCESSIVE FLOW signal to be generated on the IN SERVICE Instrument Air Dryer 00-F-104" in order to eliminate ambiguity. (licensee comment 09/11)

Question: 48 Answer: A

1 Pt(s) Given the following conditions:

- The plant is at 86% power.
- 'A' SACS Loop is supplying TACS.
- 'D' SACS pump is operating in the 'B' SACS Loop.
- 'B' SACS pump is in AUTO.
- 'A', 'C' and 'D' Service Water pumps are running.
- A guillotine rupture of the TACS piping occurs causing a Low-Low pressure in the supply accumulator.

Which of the following is the system response?

A. **The 'B' SACS pump will start, the 'B' loop SACS to TACS isolation valves will open, leaving the 'A' and 'B' SACS loops cross-connected.**

B. **The 'A' and 'C' SACS pumps will trip on low flow, the 'B' SACS pump will start, and the 'B' pump SACS to TACS isolation valve will open.**

C. **The 'B' SACS pump will start, the 'B' and 'D' SACS to TACS isolation valves will open, and the 'A' loop SACS to TACS isolation valves will shut.**

D. **All SACS pumps will trip on low SACS pump flow.**

Distracter Analysis:

A. **Correct**: HC.OP-SO.EG-0001 interlock section, closure of the E/F will cause a low loop flow which will start the auto pump and open the opposite loop valves. Result is the loops are cross-connected.
Plausible:

B. **Incorrect**: Pumps trip on high flow, not low flow.
Plausible:

C. **Incorrect**: "A" through "D" valves need a pump trip or isolation signal to close.
Plausible:

D. **Incorrect**: Pumps trip on high flow, not low flow.
Plausible:

Level: RO Exam

Lesson Plan Objective: STACS0E015
Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the Main Control Room; assess the status of the STACS or its components by evaluation of the controls/instrumentation/alarms. IAW available control room references

Source: HC Bank Q56231

Level of knowledge: Analysis

Reference(s): HC.OP-SO.EG-0001

KA: 400000.A1.03
Ability to predict and / or monitor changes in parameters associated with operating the CCWS controls including: (CFR: 41.5 / 45.5)
A1.03 CCW Pressure 2.7 2.7

Comments / Change Record:

- None

Question: 49 Answer: C

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1 Pt(s) Following a full Group I isolation at full power, HPCI received an auto initiation signal on RPV low level. During the HPCI initiation on low level, the HPCI pump flow transmitter output signal failed and was sending a constant ZERO gpm flow signal to the controller.

Which of the following describes the HPCI system response if NO operator action is taken?

The HPCI turbine will...

A. trip on overspeed and remain shutdown.

B. trip as soon as the signal is lost.

C. trip on high RPV water level.

D. operate at minimum speed.

Distracter Analysis:

A. **Incorrect**: May trip on mechanical overspeed depending on pump load but restarts after OS automatically resets.
Plausible:

B. **Incorrect**: No trip signal from loss of flow.
Plausible:

C. **Correct**: With the flow controller output sensing zero flow, the output will be maximum speed demand. HPCI will develop full flow until +54 inch high level trip is reached.

D. **Incorrect**: If the flow is sensed low, the governor will try to speed up to establish 5600 gpm.
Plausible:

Level: RO Exam

Lesson Plan Objective: HPCI00E016
(R) Given plant conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, assess the status of the HPCI System by evaluation of the controls/instrumentation/alarms, IAW control room references.

Source: HC Bank Q24866

Level of knowledge: Analysis

Reference(s): M-56, HC.OP-SO.BJ-0001

KA: 206000.A3.02
Ability to monitor automatic operations of the HIGH PRESSURE COOLANT INJECTION SYSTEM including: (CFR: 41.7 / 45.7)
A3.02 System Flow: BWR-2,3,4 3.8 3.8

Comments / Change Record:

- None

Question: 50 Answer: A

---

1 Pt(s) Which of the following correctly describes the proper method for resetting an overspeed trip of the RCIC turbine?

A. **The trip valve actuator is closed from the control room and the overspeed trip linkage must be manually relatched locally at the actuator and valve stem. The valve is then reopened from the control room.**

B. **The trip valve actuator is closed locally and the overspeed trip linkage must be manually relatched locally at the actuator and valve stem. The valve is then reopened locally.**

C. **The trip valve actuator is closed locally and the valve relatches automatically. The valve is then reopened locally.**

D. **The trip valve actuator is closed from the control room and the valve relatches automatically. The valve is then reopened from the control room.**

Distracter Analysis:

A. **Correct:** The trip valve actuator is closed from the control room and the overspeed trip linkage must be manually relatched locally at the actuator and valve stem. The valve is then reopened from the control room. Reference HC.OP-SO.BD-001 Sect 5.7

B. **Incorrect**: See justification for A.

C. **Incorrect**: See justification for A.

D. **Incorrect**: See justification for A.

Level: RO Exam

Lesson Plan Objective: RCIC00E005
Given plant conditions regarding the RCIC trip and throttle valve:

- Distinguish a mechanical overspeed trip from other trips IAW the RCIC System Lesson Plan.
- Explain how to reset a mechanical overspeed trip IAW the system operating procedure.
- Explain how the valve closes on a trip signal IAW the RCIC System Lesson Plan.

Source: HC Bank (Q54695)

Level of knowledge: Memory

Reference(s): HC.OP-SO.BD-001 Sect 5.7

KA: 217000.A4.02
Ability to manually operate and/or monitor in the control room:
(CFR: 41.7 / 45.5 to 45.8) A4.02 Turbine trip throttle valve reset 3.9
3.9

Comments / Change Record:

- None

Question: 51 Answer: C

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1 Pt(s) Which one of the following describes the purpose of the Shutdown Cooling Return Check Valves (HV-F050A/B)?

A. **Prevents bypassing SDC flow around the core through the idle SDC loop so that the RPV does NOT inadvertently heat up.**

B. **Limits RPV inventory loss in the event that the RHR pump suction valve (HV-F004) is inadvertently opened for the pump running in the SDC loop.**

C. **Limits RPV inventory loss in the event of an RHR line break on the HV-F015A(B) RHR LOOP A(B) RET TO RECIRC side of the check valve.**

D. **Ensures that all SDC flow is redirected to the LPCI injection line on a LPCI initiation since the SDC Return Check Valve will "check" closed when the LPCI injection valve (HV-F017) opens in the loop running in SDC.**

Distracter Analysis:

A. **Incorrect**: SDC core bypass is a concern through the Recirc Pump discharge and suction valves. In addition, the F015 motor operated valve is isolated on the idle loop and that valve is in series with the check valve, so there is no potential short circuit flow path.
Plausible: SDC bypass flow is a significant concern during SDC ops, and Hope Creek has Operating Experience with this problem. In addition, P&ID's and simplified drawings make it appear as though this scenario were plausible.

B. **Incorrect**: The SDC Return Check Valve would not mitigate this event since the flow path from the RPV to the Suppression Pool is on the suction side of the pump in this particular scenario. Also, with the pump running, flow will continue into the RPV through the SDC return line.
Plausible: The purpose of the check valve is to limit RPV inventory loss and there would be a *physical* path from the Suppression Pool to the RPV via the SDC return line; however, the drain down flow path is from the SDC SUCTION line (from the Recirc Loop) to the SP via the F008, F009, F006, and F004.

C. **Correct:** The valve will check closed in the event of a line break to limit RPV inventory loss. Refer to the lesson plan.

D. **Incorrect**: The RHR loop does not fully realign from SDC mode to LPCI mode. The F004 suction valves are closed and do not get an open signal. On a low level LPCI injection, the pumps wind up tripping on loss of suction path. Also, the check valve does not "check" in the LPCI flow direction.
Plausible: RHR does automatically realign from test mode to injection mode. In addition, if RHR did align from SDC to LPCI, then it would be important to redirect flow from SDC to LPCI (the motor operated SDC return valve does in fact close on a LPCI signal).

Level: RO Exam

Lesson Plan Objective: RHRSYSE003
Given a copy of P&ID M-51-1, determine each of the following operating modes of the RHR System:

a. Low Pressure Coolant Injection
b. Suppression Pool Cooling/Full Flow Test
c. Drywell Spray
d. Suppression Chamber Spray
e. Fuel Pool Cooling Assist
f. Shutdown Cooling
g. Normal Standby Alignment

Source: New

Level of knowledge: Memory

Reference(s):
Lesson Plan NOH01RHRSYSC, "RHR"

KA: 205000.G2.1.28
2.1.28 Knowledge of the purpose and function of major system components and controls. (CFR: 41.7)

Comments / Change Record:

- Added nomenclature (noun name) associated with the HV-F015A(B), and added HV- before all valve designators. (licensee 09/11)
- Capitalized "NOT" in answer "A". (licensee comment 8/27)

Question: 52 Answer: B

1 Pt(s) 125 VDC Bus B (1BD417) tripped off due to a bus fault.

From which location(s) can SRV F "Foxtrot" be opened if needed?

A. Available from the Main Control Room and Remote Shutdown Panel.

B. NOT available from any location.

C. Available Main Control Room only.

D. Available from the Remote Shutdown Panel only.

Distracter Analysis:

A. **Incorrect**: Cannot be opened from either location. See justification for answer B.
Plausible: ADS SRV's (A,B,C,D,E) each have an "A" coil and a "B" coil, so if F were mistaken for an ADS SRV, then someone might conclude that power was available from 1DD417. In addition, the valve does have control stations at both the MCR and RSP.

B. **Correct:** 1BD417 feeds all of the SRV "A" solenoids. SRV F is a non-ADS SRV, so it only has one solenoid and it is an "A" solenoid. SRV F has one control station in the MCR and one at the RSP; however, the valve cannot be opened from either location since both locations use the same "A" solenoid and there is no power available to this SRV solenoid.

C. **Incorrect**: Cannot be opened from the MCR. See justification for answer B.
Plausible: ADS SRV's (A,B,C,D,E) each have an "A" coil and a "B" coil, so if F were mistaken for an ADS SRV, then someone might conclude that power was available from 1DD417. Furthermore, all ADS SRV solenoid valves have 2 control stations in the MCR; one for the "A" solenoid and one for the "B" solenoid.

D. **Incorrect**: Cannot be opened from the RSP. See justification for answer B.
Plausible: There is a control station at the RSP for this valve. It is possible that someone could conclude that there is a "B" solenoid that could be energized via the RSP panel. Please note that ADS valves A & E does have a control station in the lower relay room that

uses a “B” solenoid powered from 1DD217, so someone could think that the F SRV might have the same setup at the RSP.

Level: RO Exam

Lesson Plan Objective: MSTEAME003
Concerning the safety relief valves; summarize, list or identify the following.

- The number and type of SRV's at Hope Creek.
- Which SRV's have an ADS function.
- Power supplies to the SRV solenoids.
- Which SRV's can be operated remotely and the location from which each of these valves can be operated.
- Purpose of the low-low set function and determine which SRV's are used for this function.
- Determine the sequence of operation of the low-low set SRV's including arming setpoints, lift points and reclose setpoints.

Source: Modified INPO Bank question (LGS)

Level of knowledge: Memory

Reference(s):
Lesson Plan

KA: 263000.K2.01
Knowledge of electrical power supplies to the following: (CFR: 41.7) K2.01 Major D.C. loads. 3.1 3.4

Comments / Change Record:

- Capitalized “NOT”. (licensee comment 07/16)

Question: 53 Answer: C

1 Pt(s) Uninterruptible Power Supply 1BD483 fails resulting in a complete loss of power to 1BJ483 distribution panel.

Which one of the following describes the feedwater system response?

A. **ALL RFPs trip due to the power interruption to DFCS.**

B. **All RFPs remain running since DFCS remains in service on the secondary power supply.**

C. **"B" RFP trips on loss of speed signal.**

D. **"B" RFP trips on loss of control power to trip solenoids.**

Distracter Analysis:

A. **Incorrect**: DFCS will remain in service on secondary DC power. This is an uninterruptible source, so there is neither a short or prolonged interruption in power. The A & C pumps remain running.
Plausible: If there was not a secondary power supply or if there was an interruption / loss of power to DFCS, then all the RFPs would trip.

B. **Incorrect**: "B" RFP will trip on loss of speed signal.
Plausible: DFCS does remain in service on secondary DC power.

C. **Correct:** A loss of AC distribution panel will result in a loss of speed pickup signal on "B" RFP, and this results in a RFP trip. DFCS (Digital FW Control System) has a secondary power supply from the DC system, so DFCS will control the 2 remaining RFPs.

D. **Incorrect**: The loss of power trip is related to a loss of DC control power NOT AC power.
Plausible: The trip is similar to the loss of speed signal trip.

Level: RO Exam

Lesson Plan Objective: FWCONTE016
Given any of the following systems, state the interrelationship between the FWLC System and that System, IAW the Feedwater Control System Lesson Plan:

- 120 VAC Non-1E Electrical Distribution
- 125 VDC Non-1E Electrical Distribution
- Main Turbine
- Recirculation System
- Rod Worth Minimizer (RWM)
- Main Steam
- Redundant Reactivity Control (RRCS)
- Hydrogen Water Chemical Injection System

Source: New

Level of knowledge: Comprehension

Reference(s):
Lesson Plan NOH04FWCONTC, "Feedwater Control System"

KA: 262002.K1.01
Knowledge of the physical connections and/or cause effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: K1.01 Feedwater level control: Plant-Specific. 2.8 3.0

Comments / Change Record:

- Deleted the word "control" from the stem question. (licensee comment 07/16)
- Added words "to trip solenoids" to distracter D. (licensee comment 08/27)

Question: 54 Answer: D

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1 Pt(s) Two minutes after the reactor scrammed at 12.5" due to a loss of two reactor feed pumps, the RO notes the CRD system flow control valve is closed but CRD system flow indication is off scale high.

Under these conditions, the RO should:

A. **Inform the CRS and have an NEO investigate the faulty valve position indication.**

B. **Trip the operating CRD pump as soon as all control rods are verified to be fully inserted to prevent pump runout.**

C. **Take manual control of the flow control valve and reduce CRD system flow to 65 gpm.**

D. **Continue with scram actions since the CRD indications are normal.**

Distracter Analysis:

A. **Incorrect**: Condition is normal. There is no need to investigate.
Plausible:

B. **Incorrect**: There is no need to trip the pump. The system's orifice is designed to prevent runout during a scram condition.
Plausible:

C. **Incorrect**: Condition is normal. There is no need to take manual control.
Plausible:

D. **Correct**: Accumulator flow will cause indication to be high, until the SCRAM is reset.

Level: RO Exam

Lesson Plan Objective: CRMECHE002
(R) Given a simplified diagram of a CRDM, explain the flowpath of water through the mechanism during the following modes of operation:

1. Insert
2. Withdrawal
3. Scram
    a. With accumulator water
    b. With reactor water
4. No Rod Motion (Cooling)

Source: HC Bank Q54980

Level of knowledge: Comprehension

Reference(s):
NOH05000006C - Control Rod Drive Hydraulics

KA: 201001.K5.02
Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD DRIVE HYDRAULIC SYSTEM: (CFR: 41.5 / 45.3) K5.02 Flow Indication 2.6 2.6.

Comments / Change Record:

- None

Question: 55 Answer: D

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1 Pt(s) The Unit is in OPCON 2 at 920 psig with the CRD Hydraulic System lined up for normal operations. Reactor pressure is lowered, and it is the only parameter that is adjusted.

Complete the following statement:

IF a control rod is given a continuous insert or withdrawal command, **THEN** the insert speed will be ______ and the withdrawal speed will be ______ than they were before reactor pressure was lowered.

A. **slower / faster**

B. **faster / slower**

C. **slower / slower**

D. **faster / faster**

Distracter Analysis:

A. **Incorrect**: The only thing that is significantly different between an insert and withdrawal command is that the drive piston is subjected to drive water pressure on the bottom of the piston, and the top side of the piston is exposed to the exhaust header (reactor pressure) to insert the rod. The reverse is true to withdraw the rod. So, this scenario results in a higher dp across the drive piston, so the speeds increase in both directions.

B. **Incorrect**: Speeds increase in both directions. (See justification for answer A).

C. **Incorrect**: The reduction in reactor pressure will result in a higher dp across the drive piston (see discussion for answer D). Therefore, insert and withdrawal speeds will be faster NOT slower.

D. **Correct:** Control rod normal insert & withdrawal speeds will be faster after the reactor pressure reduction due to a higher delta p across the drive piston. The pressure control valve is set by throttling the pressure control motor operated valve to the desired position, and the PCV does not automatically modulate to regulate the delta p. In addition, the insert and withdrawal speeds were set via needle valves that were set under previously established system conditions. In short, a reduction in reactor pressure alone will result in a higher delta p between drive pressure and reactor pressure (exhaust header), so both insert & withdrawal speeds will be faster.

Level: RO Exam

Lesson Plan Objective: ????

Source: New

Level of knowledge: Comprehension

Reference(s):
NOH04CRDHYD, "Control Rod Hydraulics"

KA: 2001003.K6.02
Knowledge of the effect that a loss or malfunction of the following will have on the CONTROL ROD AND DRIVE MECHANISM : (CFR: 41.7 / 45.7) K6.02 Reactor pressure 3.0 3.0

Comments / Change Record:
- Changed initial conditions from 100% power to OPCON 2 at 920 psig. (licensee comment)
- Replaced the word "normal" with "continuous" in the stem statement. (licensee comment 09/11)

Question: 56 Answer: C

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1 Pt(s) Recirculation loop flow mismatch limits are based on "effective core flow."

Effective core flow is determined by using:

A. **total jet pump flow.**

B. **the higher recirc loop flow and multiplying by 2.**

C. **the lower recirc loop flow and multiplying by 2.**

D. **the average of the two recirc loop flows.**

Distracter Analysis:

A. **Incorrect**:
Plausible:

B. **Incorrect**:
Plausible:

C. **Correct:** See HC.OP-ST.BB-0001.
- "Recirculation Loop Flow" refers to Jet Pump Loop Flow, not Recirculation Pump Flow.
- T/S define effective core flow as the Total Core Flow that would result if both Jet pump Loop Flows were assumed to be at the smaller value of the two flows. Example. - If Jet Pump Loop "A" Flow is 48 Mlbm/hr and Jet Pump Loop "B" Flow is 46 Mlbm/hr, the effective core flow would be 92 Mlbm/hr (46 x 2).

D. **Incorrect**:
Plausible:

Level: RO Exam

Lesson Plan Objective: IOP006E003
(R) Apply Precautions, Limitations and Notes while executing the POWER CHANGES DURING OPERATION Integrated Operating Procedure.

Source: HC Bank Q77563

Level of knowledge: Memory

Reference(s):
HC.OP-ST.BB-0001(Q) - Rev. 36 Step 5.1.4.F note

KA: 202002.A1.07
Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION FLOW CONTROL SYSTEM controls including: (CFR: 41.5 / 45.5) A1.07 Recirculation loop flow: Plant-Specific 3.1 3.1

Comments / Change Record:

- Licensee commented that the question was a difficult memory level question. Comment noted. No changes were made. This was a direct licensee bank question (licensee comment 09/11)

Question: 57 Answer: A

1 Pt(s)

A reactor startup is in progress with the reactor above the point of adding heat. RWCU was placed in blowdown operation to support level control with HV-F033 HIC-R606 DRAIN FLOW CONTROL throttled about 50% open. The operator notices that the NRHX outlet temperature is 128°F and rising.

What is the correct action to take to reduce the NRHX outlet temperature IAW HC-OP-SO.BG-0001, and what is one operational concern regarding the high temperature condition?

Throttle HV-F033 HIC-R606 in the ________ direction. ________ will automatically isolate if the NRHX outlet temperature reaches 140°F.

A. **Closed ; HV-F004, Outboard Isolation Valve**

B. **Open ; HV-F001, Inboard Isolation Valve**

C. **Closed ; HV-F001, Inboard Isolation Valve**

D. **Open ; HV-F004, Outboard Isolation Valve**

Distracter Analysis:

A. **Correct:** IAW HC-OP-SO.BG-0001

B. **Incorrect**: Opening the valve will increase blowdown flow along with the heat load on the NRHX causing NRHX outlet temperature to rise. In addition, HV-F001 will NOT isolate on high NRHX outlet temperature.
Plausible:

C. **Incorrect**: The first half is correct; however, HV-F001 will NOT isolate on high NRHX outlet temperature.
Plausible:

D. **Incorrect**: Opening the valve will increase blowdown flow along with the heat load on the NRHX causing NRHX outlet temperature to rise. The second half is correct.
Plausible:

Level: RO Exam

Lesson Plan Objective: RWCU00E018
(R) Given a set of plant conditions evaluate the effects of RWCU System blowdown operation on the RHX and NRHX's IAW the RWCU System Lesson Plan.

Source: New

Level of knowledge: Analysis

Reference(s):

- HC-OP-SO.BG-0001 rev. 40 Section 3.3.1 & Step 5.4.8
- HC.OP-IO.ZZ-003 rev. 76 Section 3.3.5

KA: 204000.A2.14
Ability to (a) predict the impacts of the following on the REACTOR WATER CLEANUP SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.14 System high temperature 3.2 3.2

Comments / Change Record:

- None

Question: 58 Answer: A

1 Pt(s) Given the following:

- OPCON 1 at 100% power.
- All control rods are fully withdrawn.
- The Reactor Operator begins HC.OP-ST.SF-003, "RPS Manual SCRAM Test – Weekly".
- The Reactor Operator arms and presses the RPS TRIP SYS A1 MAN INIT PB.
- Coincidentally, the PILOT SCRAM VALVE TRIP ACTUATOR LOGIC Group 4B SOLENOIDS LOGIC B NORMAL light extinguishes at 10C651C. This was NOT caused by the previous Operator action.

Which of the following describes the indications on the full core display based solely on the INITIAL RPS response?

A. **Control Rod Groups 1, 2, & 3 RED lamps are lit, and Control Rod Group 4 GREEN, BLUE, & AMBER lamps are lit.**

B. **All Control Rods have RED lamps lit.**

C. **All Control Rods have GREEN, BLUE, & AMBER lamps lit.**

D. **Control Rod Groups 1 & 4 GREEN, BLUE, & AMBER lamps are lit, and Control Rod Groups 2 & 3 have RED lamps lit.**

Distracter Analysis:

A. **Correct:** The Operator's action causes an "A" side half scram and all "A" side scram pilot solenoids de-energize. No rod motion occurs. Now, the Group 4B solenoid logic status light extinguishes indicating a loss of power to "B" side scram pilot solenoids for all control rods in Group 4 only. So, the "A" and "B" scram pilot solenoid valves are both de-energized on all control rods in Group 4, and this vents air from all scram valves for control rods in Group 4 ONLY. The Full Core Display will show RED (full out indication) for rods in Groups 1, 2, & 3, and Group 4 control rods will display GREEN (full in), BLUE (scram valves open), and AMBER (rod drift).

B. **Incorrect**: RED indicates full out position. Group 4 control rods went full in, so this answer is incorrect.

Plausible: One could conclude that sufficient logic did not complete to cause any of the rods to scram.

C. **Incorrect**: This would be an indication of a full scram; however, this problem only caused Group 4 rods to scram in.
Plausible: It is possible that someone could conclude sufficient logic made up to cause a full core scram.

D. **Incorrect**: As stated above, scram logic de-energized to the extent that ONLY the Group 4 control rods scrammed.
Plausible: The nomenclature could lead a candidate to conclude that the control rods associated with Groups 1 & 4 scrammed (A1 manual pushbutton was depressed and the Group 4B logic power failure, i.e. RPS A & B de-energized on Groups 1 & 4). The justification for the correct answer describes how the logic actually works.

Level: RO Exam

Lesson Plan Objective: MANCONE002

Source: New

Level of knowledge: Analysis

Reference(s):
Lesson Plan NOH01RPS00C, "Reactor Protection System"
Lesson Plan NOH04MANCONC, "Reactor Manual Control System"
HC.OP-ST.SF-0003, "RPS Manual Scram Test – Weekly"

KA: 214000.A3.01
Ability to monitor automatic operations of the ROD POSITION INFORMATION SYSTEM including: (CFR: 41.7 / 45.7) A3.01 Full core display 3.4 3.3

Comments / Change Record:

- Deleted the words "best" and "expected" from the stem question.
- Added words to the stem question "based solely on the INITIAL RPS response". (licensee comment 08/28)

Question: 59 Answer: C

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1 Pt(s) Given the following conditions:

- A LOCA has occurred in the Drywell
- RPV level is +20 inches and is steady
- Suppression Chamber pressure is 7.3 psig and rising slowly
- The CRS directs "B" RHR be placed in Suppression Chamber Spray

Which of the following operator actions and plant conditions are required to establish flow through the Suppression Chamber Spray valve(s) BC-HV-F027B?

The Suppression Chamber Spray valve(S) BC-HV-F027B Auto _____(1)______ Overide must be pressed and the ______(2)_______ must be ______(3)_______.

A. **(1) Open, (2) LPCI initiation signal, (3) reset**

B. **(1) Close, (2) LPCI initiation signal, (3) reset**

C. **(1) Close, (2) High Drywell pressure signal, (3) present**

D. **(1) Open, (2) High Drywell pressure signal, (3) present**

Distracter Analysis:

A. **Incorrect**: All three selections are incorrect.
Plausible:
B. **Incorrect**: Items 2 & 3 are incorrect.
Plausible:
C. **Correct**
D. **Incorrect**: Item 1 is incorrect.
Plausible:

Level: RO Exam

Lesson Plan Objective: RHRSYSE014

Source: HC Bank Q56222 (Audit 1999)

Level of knowledge: Comprehension

Reference(s): HC.OP-SO.BC-0001

KA: 230000.A4.02
Ability to manually operate and/or monitor in the control room:
(CFR: 41.7 / 45.5 to 45.8) A4.02 Spray Valves 3.8 3.6

Comments / Change Record:

- None

Question: 60 Answer: D

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1 Pt(s) The unit is shutdown in OPCON 4.

SELECT the system capable of providing emergency makeup to the spent fuel pool that can be completely aligned from the control room.

A. Safety Auxiliary Cooling System.

B. Fire protection.

C. Residual Heat Removal.

D. Service Water System.

Distracter Analysis:

A. **Incorrect**: Not able to supply emergency fill.
Plausible:
B. **Incorrect**: Cannot be aligned for the control room.
Plausible:
C. **Incorrect**: Not available until Refueling operations via LPCI from Suppression Pool or manual valve operation from the field.
Plausible:
D. **Correct:** Can be aligned via valves form the control room.

Level: RO Exam

Lesson Plan Objective: FPCC00E008

Source: HC Bank Q60993

Level of knowledge: Memory

Reference(s): M-10-1 and M-53-1

KA: 233000.G2.1.30
Ability to locate and operate components / including local controls. (CFR: 41.7 / 45.7) 3.9 3.4.

Comments / Change Record:

- None

Question: 61 Answer: C

1 Pt(s) Which of the following conditions will result in an automatic closure of the Main Steam Supply Valve (AB-HV-2016A/B) to the steam jet air ejectors?

A. **Low offgas flow to the third stage air ejector for thirty seconds and the first stage air ejector suction valves (CG-HV-1968A1,A2,A3/B1,B2,B3) are open.**

B. **Low offgas flow to the third stage air ejector for thirty seconds and the third stage air ejector suction valve (CG-HV-1956A/B) is open.**

C. **Low main steam flow to the third stage air ejector for thirty seconds and the first stage air ejector suction valves (CG-HV-1968A1,A2,A3/B1,B2,B3) are open.**

D. **Low main steam flow to the third stage air ejector for thirty seconds and the third stage air ejector suction valve (CG-HV-1956A/B) is open.**

Distracter Analysis:

A. **Incorrect**: Low offgas flow is incorrect.
Plausible:
B. **Incorrect**: Low offgas flow is incorrect.
Plausible:
C. **Correct:** IAW HC.OP-SO.CG-0001.
D. **Incorrect**: Third stage is incorrect.
Plausible:

Level: RO Exam

Lesson Plan Objective: AIRREME011

Source: HC Bank Q53605

Level of knowledge: Memory

Reference(s): HC.OP-SO.CG-0001

KA: 239001.K1.07
Knowledge of the physical connections and/or cause effect relationships between MAIN AND REHEAT STEAM SYSTEM and the following: (CFR: 41.2 to 41.9 / 45.7 to 45.8) K1.07 Offgas system 3.1 3.1

Comments / Change Record:

- None

Question: 62 Answer: D

1 Pt(s) The plant is operating at 97% power limited by turbine back pressure. The "B" primary condensate pump trips. Which of the following is the correct automatic response of the plant?

A. **RFP's are limited to 4961 rpm; Recirculation System receives an intermediate runback signal**

B. **RFP's are limited to 4961 rpm; Recirculation System is unaffected.**

C. **RFP's are limited to 4816 rpm; Recirculation System is unaffected.**

D. **RFP's are limited to 4816 rpm; Recirculation System receives a full runback signal.**

Distracter Analysis:

A. **Incorrect**: 4816 RPM from DFCS and full runback on Recirc see HC.OP-SO.BB-0002 and HC.OP-SO.AE-0001.
Plausible: 4961 rpm is the intermediate runback for a secondary condensate pump trip.

B. **Incorrect**: 4816 RPM from DFCS and full runback on Recirc see HC.OP-SO.BB-0002.
Plausible: 4961 rpm is the intermediate runback for a secondary condensate pump trip.

C. **Incorrect**: Full runback on Recirc; see HC.OP-SO.BB-0002.
Plausible:

D. **Correct**: IAW HC.OP-SO.BB-0002 and HC.OP-SO.AE-0001 Rev 48 Section 3.3.3 & 3.3.4.

Level: RO Exam

Lesson Plan Objective: FWCONTE015
(R) From memory, describe the three possible RFP runback signals including conditions, setpoints and time delays if applicable, IAW the Feedwater Control System Lesson Plan.

Source: HC Bank Q56801

Level of knowledge: Memory

Reference(s):

- HC.OP-SO.AE-0001 Rev. 48 Section 3.3.3 & 3.3.4 ***(Revision with RPM change as provided by Archie Faulkner)***
- HC.OP-SO.BB-0002

KA: 259001.K4.10
Knowledge of electrical power supplies to the following: (CFR: 41.7) Feedpump runbacks: Plant-Specific 3.1 3.4

Comments / Change Record:

- None

Question: 63 Answer: A

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1 Pt(s) The plant is in OPCON 1 at 20% power with the following offgas system indications:

- Feed Gas Cooler Condenser Outlet Temperature is 250°F
- Preheater Outlet Temperature is 300°F
- Recombiner Outlet Temperature is 750°F

Which one of the following describes the status of Main Condenser backpressure and the availability of the Mechanical Vacuum Pump (MVP)?

Under the current conditions, Main Condenser backpressure.....

A. will rise. The MVP may NOT be placed in service.

B. will rise. The MVP may be placed in service if needed.

C. will remain stable. The MVP may be placed in service if needed.

D. will remain stable. The MVP may NOT be placed in service.

Distracter Analysis:

A. **Correct:** The offgas train will isolate when Feed Gas Cooler Condenser Outlet Temperature reaches 200°F resulting in a rise in Main Condenser backpressure. The MVP may NOT be placed in service with reactor power above 5%.

B. **Incorrect**: The MVP may NOT be placed in service with reactor power above 5%.
Plausible:

C. **Incorrect**: The offgas train will isolate when Feed Gas Cooler Condenser Outlet Temperature reaches 200°F resulting in a rise in Main Condenser backpressure. The MVP may NOT be placed in service with reactor power above 5%.
Plausible:

D. **Incorrect**: The offgas train will isolate when Feed Gas Cooler Condenser Outlet Temperature reaches 200°F resulting in a rise in Main Condenser backpressure.
Plausible:

Level: RO Exam

Lesson Plan Objective: GASRW0E013
(R) Concerning Recombiner Trips/Isolations, determine the following IAW available control room references:
a. The signals which will cause a recombiner isolation
b. Valves affected by an isolation signal

Source: New

Level of knowledge: Analysis

Reference(s):
HC.OP-AB.BOP-0006 rev. 9

KA: 271000.K3.01
Knowledge of the effect that a loss or malfunction of the OFFGAS SYSTEM will have on following: (CFR: 41.5 / 45.3) K3.01 Condenser vacuum 3.5 3.5

Comments / Change Record:

- Reclassified cognitive level from memory to analysis. The question tests knowledge of isolation setpoints as well as the ability to predict the impact on main condenser vacuum. Furthermore, the question tests knowledge of the permissibility of using a system to mitigate the problem. (NRC Exam Author 09/14)

Question: 64 Answer: C

1 Pt(s) Given the following:

- The plant is operating at 100 percent power
- A Loss of Offsite power occurs coincident with a LOCA
- Drywell pressure is 5 psig and rising
- "A" Emergency Diesel Generator fails to start

SELECT the total FRVS recirculation flow 3 minutes after the LOCA/LOP began. Assume NO operator action.

A. **0 cfm**

B. **90,000 cfm**

C. **120,000 cfm**

D. **180,000 cfm**

Distracter Analysis:

A. **Incorrect**: 4 fans start for a total of 120,000 cfm.
Plausible: Would be true if there was no LOCA initiation signal.

B. **Incorrect**: 4 fans start for a total of 120,000 cfm.
Plausible: Would be true if 3 fans started. Need to know power supplies and system response to an initiation signal.

C. **Correct:** "A" EDG powers "A" & "E" FRVS Recirculation fans. B, C, D, & F will start on the high drywell pressure signal. Each fan develops 30,000 cfm recirculation flow, so 4 X 30,000 cfm equals a total of 120,000 cfm.

D. **Incorrect**: Only 4 fans start for a total of 120,000 cfm.
Plausible: Would be true if all fans had power and NO operator action. Normally 6 fans start for 180,000 cfm and then operators secure 2 fans.

Level: RO Exam

Lesson Plan Objective: SECCONE012
(R) Given a set of conditions and a drawing of the controls, instrumentation and/or alarms located in the main control room, identify the status of the Secondary Containment by evaluation of the controls/ instrumentation/alarms.

Source: Mod HC Bank Q68889

Level of knowledge: Analysis

Reference(s): HC.OP-SO.GU-0001

KA: 288000.K4.02
Knowledge of PLANT VENTILATION SYSTEMS design feature(s) and/or interlocks which provide for the following: (CFR: 41.7) K4.02 Secondary containment isolation. 3.7 3.8

Comments / Change Record:

- None

Question: 65 Answer: A

1 Pt(s) WHICH of the following describes the setpoint when the Reactor Building-to-Torus Vacuum Breakers butterfly isolation valve opens and the valve type?

When Reactor Building pressure exceeds the Suppression Chamber pressure by:

A. **0.25 psid, a pneumatic actuator opens the butterfly isolation valve.**

B. **0.50 psid, a pneumatic actuator opens the butterfly isolation valve.**

C. **0.50 psid, a motor operator opens the butterfly isolation valve.**

D. **0.25 psid, a motor operator opens the butterfly isolation valve.**

Distracter Analysis:

A. **Correct:** T.S surveillance 4.6.4.2.b.2.a) states valve opening not to exceed 0.25 psid. M-57-1 shows the butterfly isolation valves have pneumatic actuators

B. **Incorrect:** The valve opening setpoint is 0.25 psid.
Plausible:

C. **Incorrect:** Valve opening setpoint is 0.25 psid; the valve has a pneumatic actuator.
Plausible:

D. **Incorrect:** The valve has a pneumatic actuator.
Plausible:

Level: RO Exam

Lesson Plan Objective: PRICONE003
Summarize the basic construction and function of the following Primary Containment components:
Vacuum Breakers, etc.

Source: HC Bank Q55764

Level of knowledge: Memory

Reference(s): M-57-1 & Lesson Plan

KA: 290001.K5.01
Knowledge of the operational implications of the following concepts as they apply to SECONDARY CONTAINMENT : (CFR: 41.5 / 45.3) K5.01 Vacuum breaker operation: BWR-4. 3.3* 3.4*

Comments / Change Record:

- None

Question: 66 Answer: B

1 Pt(s) The reactor is at 90% power during a control rod exchange. The intermediate control rod configuration resulted in operations with MCPR equal to the limit specified in the COLR.

What MUST be done prior to withdrawing the next control rod?

A. Core flow must be reduced until the rod line is lowered.

B. A RBM Channel Functional Test must be performed.

C. The reactor engineer must revise the RWM rod sequence.

D. The Pressure Regulator setpoint must be lowered.

Distracter Analysis:

A. **Incorrect**: This may cause the Thermal Limit to exceed the COLR setpoint, since MCPR is flow dependent.
Plausible: MCPR is flow dependent.

B. **Correct:** IAW HC.OP-IO.ZZ-0006 Caution 5.1.6 and Technical Specification 4.1.4.3.b.

C. **Incorrect**: This is an option to correct the pattern but it is not a requirement, but the TS specifically require a RBM functional prior to rod withdrawal while on a limiting pattern.
Plausible: This is an option to correct the problem.

D. **Incorrect**: Changing the regulator setpoint will change void concentration and a change in the void concentration may cause the thermal limit to lower or rise based on rod pattern. In addition, TS specifically requires a RBM functional prior to the next rod withdrawal while on a limiting pattern.
Plausible: Could correct the problem depending on the rod pattern.

Level: RO Exam

Lesson Plan Objective: IOP006E005
(R) Analyze plant conditions and parameters to determine if plant operation is in accordance with the POWER CHANGES DURING OPERATION Integrated Operating Procedure, supporting System Operating Procedures and Technical Specifications.

Source: HC Bank Q57125

Level of knowledge: Analysis

Reference(s): HC.OP-IO.ZZ-0006 rev 37 - step 5.1.6 Caution
Prior to control rod withdrawal, if the reactor is operating on a LIMITING CONTROL ROD PATTERN, then Tech Spec 4.1.4.3.b shall be complied with.

- Technical Specifications define a LIMITING CONTROL ROD PATTERN as a pattern that results in the core being on a thermal hydraulic limit; i.e., operating on a limiting value for APLHGR, LHGR or MCPR.
- Technical Specification 4.1.4.3.b requires that a Channel Functional Test for the Rod Block Monitor be performed prior to control rod withdrawal when the reactor is operating on a Limiting Control Rod Pattern.

KA: G2.1.12
2.1.12 Ability to apply technical specifications for a system. (CFR: 43.2 / 43.5 / 45.3) IMPORTANCE RO 2.9 SRO 4.0

Comments / Change Record:

- Licensee requested that this question be classified as SRO only; however, the request was denied since it is based on a valid RO K/A with an RO Importance Value of 2.9. (licensee comment 09/11)

Question: 67 Answer: A

1 Pt(s) Given the following conditions:

- Reactor pressure is greater than 785 psig.
- Core flow is greater than 10% of rated flow.
- Both Reactor Recirculation Pumps are in service.

Which of the following is a violation of a Thermal Power Safety Limit as defined in Technical Specifications?

A. **MCPR less than 1.06**

B. **MCPR equal to 1.06**

C. **MCPR equal to 1.08**

D. **MCPR greater than 1.08**

Distracter Analysis:

A. **Correct:** Refer to Technical Specification Safety Limit 2.1.2. Any MCPR value less than 1.06 is a violation for two loop operation for the given pressure and flow conditions.

B. **Incorrect**: Per Safety Limit 2.1.2, MCPR shall be greater than **OR equal to 1.06** for two loop operations, so this is NOT a Safety Limit violation.
Plausible: 1.06 is the lowest value of MCPR for two loop operation that is NOT a violation of the Safety Limit. The distracter also tests candidate knowledge of the "conservative direction" for the MCPR value.

C. **Incorrect**: Per Safety Limit 2.1.2, MCPR shall be greater than **or equal to 1.06** for two loop operations, so this is NOT a Safety Limit violation.
Plausible: 1.08 is the lowest value of MCPR for **single loop** operation that is NOT a violation of the Safety Limit. The distracter also tests candidate knowledge of the "conservative direction" for the MCPR value.

D. **Incorrect**: Per Safety Limit 2.1.2, MCPR shall be greater than **or equal to 1.06** for two loop operations, so this is NOT a Safety Limit violation.
Plausible: 1.08 is the lowest value of MCPR for **single loop** operation that is NOT a violation of the Safety Limit. For the given

pressure and flow conditions, MCPR must be maintained **greater than OR equal to 1.08 for *single loop* operations**. The distracter tests the candidate's knowledge of the difference between the limits for single loop versus two loop operation as well as knowledge of the "conservative direction" for the MCPR value.

Level: RO Exam

Lesson Plan Objective: TECSPCE001
State the four (4) Safety Limits in terms of conditions.

Source: New

Level of knowledge: Memory

Reference(s): Technical Specification Safety Limit 2.1.2

KA: G2.1.10
Knowledge of conditions and limitations in the facility license.
(CFR: 43.1 / 45.13) IMPORTANCE RO 2.7 SRO 3.9

Comments / Change Record:

- This a replacement question. The previous question was related to operation above the licensed power level. During the validation process, RIS-2007-21 was issued communicating that licensee's must not exceed the licensed power level. A review of the previous exam question revealed that the question was in conflict with agency guidance; therefore, the previous question was determined to be not valid and was rejected as part of the exam validation process. (NRC exam author 09/11)

Question: 68 Answer: A

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1 Pt(s) You and the Rx Building NEO have been directed to reposition and independently verify EG-V541, RHR HX BE205 SACS RTN ISLN VLV, located in the B RHR Heat Exchanger Room. The Heat Exchanger Room is posted as a "Locked High Rad Area."

Which of the following is an acceptable means for Independent Verification?

A. **Observe change in SACS flow using CRIDS and hardwire indications in the Main Control Room.**

B. **Verification using remote position indication in the Main Control Room.**

C. **NO Independent Verification is required due to high cumulative radiation exposure.**

D. **After initial positioning and locking of the valve, remove the lock count the turns to close, reposition valve and relock.**

Distracter Analysis:

A. **Correct:** IAW SH.OP-AP.ZZ-103, rev 12, attachment 12, section 1.7.b.

B. **Incorrect**: Remote position indication is not available for this valve.
Plausible:

C. **Incorrect**: Some form of IV is always required.
Plausible:

D. **Incorrect**: high dose would dictate other means for verification, improper verification of a locked valve this would be a dual verification.
Plausible:

Level: RO Exam

Lesson Plan Objective: ADMPRO103CE001
From Memory determine the requirements for Independent Verification IAW SH.OP-AP.ZZ-0103 and HU-AA-101.

Source: HC Bank Q56288

Level of knowledge: Memory

Reference(s): SH.OP-AP.ZZ-103, rev 12, Attachment 12, Section 1.7.b

KA: G2.1.29
2.1.29 Knowledge of how to conduct and verify valve lineups.
(CFR: 41.10 / 45.1 / 45.12) IMPORTANCE RO 3.4 SRO 3.3

Comments / Change Record:

- Corrected capitalization errors: "hardware" in answer "A", "indication" in answer "B", and "initial" in answer "D". (licensee comment 09/11)

Question: 69 Answer: B

1 Pt(s) Given the following conditions:

- Reactor power is 30%
- "D" Diesel Generator is paralleled with the bus for surveillance testing
- "D" DG load is 2200KW
- A system malfunction causes an inadvertent initiation signal in the "D" Core Spray logic

Which statement below describes the response of the "D" DG to the conditions above?

A. The Normal bus supply breaker trips open and the diesel generator supplies the 10A404 bus.

B. The generator output breaker will trip open but the diesel remains running.

C. The diesel generator remains in parallel operation.

D. The generator output breaker will trip. The diesel shuts down and must be manually restarted.

Distracter Analysis:

A. **Incorrect**: The EDG breaker will trip, and the normal supply breaker will carry the bus.
Plausible:

B. **Correct:** The following response occurs when a LOCA signal is received while the DG is operating in Test: **the DG Output Breaker Trips, the DG remains running**, the voltage regulator shifts to ISOC, and the Test Lockout Relay trips are disabled. **If offsite power is available, the bus remains powered from the normal or alternate feed.** If breaker closure permissives are met, the DG breaker will close to restore power.

C. **Incorrect**: The EDG breaker will trip, the EDG will remain running, and the governor and regulator will shift to isochronous.
Plausible:

D. **Incorrect**: The EDG breaker will trip; however, the EDG will remain running.
Plausible:

Level: RO Exam

Lesson Plan Objective: 1EAC00E010
(R) From memory, summarize/identify the response of the diesel generators when running in the test mode and a LOP/LOCA signal is received.

Source: HC Bank Q55084

Level of knowledge: Memory

Reference(s):
Lesson Plan

KA: G2.2.12
2.2.12 Knowledge of surveillance procedures. (CFR: 41.10 / 45.13)
IMPORTANCE RO 3.0 SRO 3.4

Comments / Change Record:
- None

Question: 70 Answer: D

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1 Pt(s) Given the following initial conditions:

- The Unit is in OPCON 1 at 100% power.
- HPCI is currently operable.
- 1AVH-209 HPCI Room Cooler control switch is in "AUTO LEAD".
- 1BVH-209 HPCI Room Cooler control switch is in "AUTO"

Later in the shift,

- A Maintenance Electrician reports to the Control Room and requests permission to place 1BVH-209 HPCI Room Cooler control switch in "STOP" for about 30 minutes to take voltage readings on the control switch contacts.
- The Electrician states that he will remain near the switch while performing his work and that he can restore his work and place the control switch back to "AUTO" in less than 15 minutes in the event of a HPCI start.
- The CRS gives the Electrician permission to perform the work.
- The Electrician places the 1BVH-209 HPCI Room Cooler control switch in "STOP" at 10:00 (the current time).

What is the status of HPCI?

A. **HPCI is INOPERABLE since BOTH Room Coolers are NOT operable.**

B. **HPCI remains operable since Room Coolers are NOT required to support HPCI operability.**

C. **HPCI remains operable since BOTH Room Coolers are operable to support HPCI.**

D. **HPCI remains operable since ONE Room Cooler is operable to support HPCI.**

Distracter Analysis:

A. **Incorrect**: HPCI is operable since 1AVH-209 is unaffected by the work and only one cooler is required to be operable to support HPCI operability,

Plausible: Would be true if both coolers were required to support HPCI operability.

B. **Incorrect**: At least one room cooler is required to support HPCI operability.
Plausible: Some plants have performed analyses to prove that room coolers are not required for HPCI operability, so the answer is plausible.

C. **Incorrect**: 1BVH-209 is NOT operable. Credit can NOT be taken for manual action to restore the cooler in this case since there was no mention of an operability determination to support manual action in the stem. In addition, it is not likely that the Electrician would qualify as a "dedicated" operator for compensatory action purposes.
Plausible: Would be correct if the credit were taken for manual action. This distracter might be attractive to someone believing that both coolers are required for HPCI operability.

D. **Correct:** One cooler is required to be operable to support HPCI in accordance with HC.OP-AP.ZZ-0108. 1AVH-209 is unaffected by the work and remains operable; therefore, HPCI is operable.

Level: RO Exam

Lesson Plan Objective: HPCI00E006
(R) From memory, summarize the interrelationship(s) between the HPCI System and any of the following IAW control room references: RBVS, etc.

Source: New

Level of knowledge: Analysis

Reference(s):

- HC.OP-AP.ZZ-0108 rev. 28

KA: G2.2.24
2.2.24 Ability to analyze the affect of maintenance activities on LCO status. (CFR: 43.2 / 45.13) IMPORTANCE RO 2.6 SRO 3.8

Comments / Change Record:

- Licensee requested that this question be classified as SRO only; however, the request was denied since it is based on a valid RO K/A with an RO Importance Value of 2.6. (licensee comment 09/11)

Question: 71 Answer: B

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1 Pt(s) Given the following conditions:

- The Hope Creek Reactor is being shutdown to comply with Tech Specs
- Rx power is 15% by APRMs
- RWM Group 34 has 4 of its 18 rods inserted to the target position of Notch 12
- A rod is selected from RWM Group 16

The selected control rod can:

A. NOT be moved via RMCS; OUT OF SEQUENCE will be displayed on the RWM.

B. NOT be moved via RMCS; SELECT ERROR will be displayed on the RWM.

C. be moved via RMCS; SELECT ERROR will be displayed on the RWM.

D. be moved via RMCS; WITHDRAW ERROR will be displayed on the RWM.

Distracter Analysis:

A. Incorrect: OUT OF SEQUENCE is displayed under the following conditions: (1) >1 IE, (2) >1 WE, (3) 1 IE and 1 WE, or (4) 1 IE or WE that is >2 notches. In this case there are no INSERT or WITHDRAW errors, only a SELECT ERROR.
Plausible:

B. Correct: Selection of a control rod that is <u>not</u> in the currently latched RWM Step (Pull Sheet Group) will result in a SELECT ERROR (SE), which results in an INSERT BLOCK (IB) and WITHDRAW (WB) block for that control rod.

C. Incorrect: The rod cannot be moved due to INSERT <u>and</u> WITHDRAW blocks being applied whenever there is a SELECT ERROR.
Plausible:

D. Incorrect: The rod cannot be moved due to INSERT <u>and</u> WITHDRAW blocks being applied whenever there is a SELECT ERROR.
Plausible:

Level: RO Exam

Lesson Plan Objective: RODMINE003
(R) Given a labeled drawing of, or access to, the RWM Operator Display on 10C651, or the RWM Computer Display (in the Computer Room):
a. Explain the function of each indicator.
b. Assess plant conditions, which will cause the indicator to light or extinguish.
c. Determine the effect of each control on the Rod Worth Minimizer.
d. Assess plant conditions or permissives required for the control switches/pushbuttons to perform their intended functions.
Source: HC Bank Q54169

Level of knowledge: Memory

Reference(s):
- HC.OP-IO.ZZ-0003, Section 5.2, Approach to Criticality
- HC.OP-SO.SF-0003, Section 5.3, Operations with Insert/Withdraw Errors

KA: G2.2.33
2.2.33 Knowledge of control rod programming. (CFR: 43.6)
IMPORTANCE RO 2.5 SRO 2.9

Comments / Change Record:
- None

Question: 72 Answer: B

1 Pt(s) Given the following conditions:

- Reactor power is 100%.
- The HWCI System is in service.
- Maintenance must be performed close to the Main Steam Lines.

Which of the following describes the operation of the HWCI System regarding ALARA?

Per HC.OP-IO.ZZ-0006, Power Changes during operation; the Hydrogen and Oxygen Chemistry (HWCI) System should be……

A. shutdown for at least 8 hours prior to maintenance to reduce plant background radiation AND the shutdown should be limited to less than 24 hours since IGSCC may reinitiate after 24 hours.

B. shutdown to reduce plant background radiation during maintenance AND the shutdown should be limited to less than 8 hours since IGSCC may reinitiate after 8 hours.

C. left in service continuously to prevent IGSCC because operation of HWCI does NOT affect radiation levels.

D. left in service continuously to reduce plant background radiation AND to prevent IGSCC.

Distracter Analysis:

A. **Incorrect**: The HWCI outages should be infrequent, and as short as possible, and preferably shorter than 8 hours. IGSCC may reinitiate after 8 hours without HWCI out of service. In addition, the dose rate drops of as soon as H2 water chemistry is secured so there is no need to wait 8 hours.
Plausible: Shutting down HWCI will reduce rad levels. In addition, the procedure states that HWCI outages should be LIMITED to 8 hours, so it is plausible for someone to incorrectly recall the significance of the 8 hour time frame.

B. **Correct:** IAW HC.OP-IO.ZZ-0006 rev 37 section 3.2.6 states Background Radiation exposure within the turbine and reactor buildings will rise because of Hydrogen Water Chemistry (HWCI) system operation. If ALARA conditions warrant, HWCI should be shut down to perform maintenance or operations. The HWCI outages should be infrequent and as short as possible and preferably shorter than 8 hours. Intergrannular stress corrosion crack (IGSCC) growth may reinitiate after about 8 hours without HWCI.

C. **Incorrect**: If ALARA conditions warrant, the HWCI System should be shutdown to perform maintenance or operations.
Plausible: In order to prevent IGSCC, it is desirable to leave HWCI in service.

D. **Incorrect**: Background radiation exposure within the turbine and reactor buildings will increase because of Hydrogen and Oxygen Chemistry (HWCI) System Operation
Plausible: It is plausible that someone could have the misconception that the purpose of HWCI is to reduce background rad levels.

Level: RO Exam

Lesson Plan Objective: IOP006E002
(R) Explain the Precautions, Limitations and Notes listed in the POWER CHANGES DURING OPERATION Integrated Operating Procedure.

Source: HC Bank Q57123

Level of knowledge: Memory

Reference(s): HC.OP-IO.ZZ-0006 rev 37 section 3.2.6

KA: G2.3.10
2.3.10 Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure. (CFR: 43.4 / 45.10)
IMPORTANCE RO 2.9 SRO 3.3

Comments / Change Record:

- Added words to answers / distracters to improve plausibility and psychometric balance. (NRC reviewer comments 08/23)
- Added a bullet in the stem to state the HWCI is in service. Revised the last bullet in the stem to replace "is being performed" with "must be performed", and replaced the word "adjacent" with "close". Revised all answers to ensure that each answer has an element dealing with radiation and

an element dealing with IGSCC. Corrected an error in the correct answer "B" so that the answer states that background radiation levels will be lowered when HWCI is shutdown. The previous version implied that rad levels would rise. (licensee comments 09/11)

Question: 73 Answer: B

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1 Pt(s) A new 25 year old employee with NO prior dose is limited to an initial whole body dose control level of _________ mrem/yr TEDE for the current year, AND the dose control level may be increased to as high as ________mrem/yr TEDE for the first extension, with the proper approvals, PRIOR to exceeding the initial whole body dose control level.

A. 2000 ; 2750

B. 2000 ; 3000

C. 4000 ; 4750

D. 4000 ; 5000

Distracter Analysis:

A. **Incorrect**: Initial level is correct, but the dose may be extended to as high as 3000 mrem/yr TEDE.
Plausible: Initial level is correct. An extension could be granted to 2750; however, the first extension can go as high as 3000.

B. **Correct:** IAW NC.NA-AP-ZZ-0024 rev. 13 Attachment 1.

C. **Incorrect**: The initial is 2000 mrem/yr TEDE and the first extension is to as high as 3000 mrem/yr TEDE.
Plausible: Correct values after the SECOND extension.

D. **Incorrect**: The initial is 2000 mrem/yr TEDE and the first extension is to as high as 3000 mrem/yr TEDE.
Plausible: 4000 is correct control level after the SECOND extension and 5000 is the federal annual limit.

Level: RO Exam

Lesson Plan Objective: NOH04ADM024C-04
Given a set of exposure conditions
Identify the personnel responsible for approval of the following dose extension:

a. Yearly Dose Extension
b. Declared Pregnant Women Dose Extension
c. Lifetime Dose Extension

IAW NC.NA-AP.ZZ-0024

Source: New

Level of knowledge: Memory

Reference(s): NC.NA-AP.ZZ-0024 rev. 13 Attachment 1

KA: G2.3.4
2.3.4 Knowledge of radiation exposure limits and contamination control / including permissible levels in excess of those authorized. (CFR: 43.4 / 45.10) IMPORTANCE RO 2.5 SRO 3.1

Comments / Change Record:

- None

Question: 74 Answer: D

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1 Pt(s) A transient is in progress and you need to communicate with the crew regarding plant status, set priorities, and discuss what is coming up next.

What type of verbal communication is appropriate for this situation IAW OP-AA-104-101?

A. **Pre-job Brief**

B. **Heightened Level of Awareness (HLA) Brief**

C. **Crew Update**

D. **Summary Brief**

Distracter Analysis:

A. **Incorrect**: Used during normal non transient conditions.
Plausible:

B. **Incorrect**: Used during normal non transient conditions.
Plausible:

C. **Incorrect**: Used for quick communications (10 to 15 second duration) for procedure transitions, announcements of major plant status changes, etc.
Plausible:

D. **Correct:** IAW OP-AA-104-101. Used to brief the crew during transient conditions (1 to 3 minute duration) for the types of communications given in the stem.

Level: RO Exam

Lesson Plan Objective: ????

Source: New

Level of knowledge: Memory

Reference(s): OP-AA-104-101

KA: G2.4.15
2.4.15 Knowledge of communications procedures associated with EOP implementation. (CFR: 41.10 / 45.13) IMPORTANCE RO 3.0 SRO 3.5

Comments / Change Record:

- None

Question: 75 Answer: C

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1 Pt(s) Given the following:

- The plant is operating at 15% power, startup in progress
- A SINGLE control rod SCRAM just occurred as a result of a troubleshooting error while investigating a SCRAM test switch problem.

Which of the following groups of alarms are consistent with this condition?

A. **CRD ACCUM TROUBLE (C6-D4)**
ROD DRIFT (C6-E3)
SCRAM AIR HEADER LOW PRESSURE (CRIDS A3016)

B. **ROD DRIFT (C6-E3)**
ROD OUT MOTION BLOCK (C6-D3)
SCRAM AIR HEADER LOW PRESSURE (CRIDS A3016)

C. **CRD ACCUM TROUBLE (C6-D4)**
ROD DRIFT (C6-E3)
ROD OUT MOTION BLOCK (C6-D3)

D. **CRD ACCUM TROUBLE (C6-D4)**
ROD OVERTRAVEL (C6-F3)
ROD OUT MOTION BLOCK (C6-D3)

Distracter Analysis:

A. **Incorrect**: SCRAM AIR HEADER LOW PRESSURE will not come in on a single rod scram.
Plausible: 2 out of 3 are correct. SCRAM AIR HEADER LOW PRESSURE comes in on a full scram.

B. **Incorrect**: SCRAM AIR HEADER LOW PRESSURE will not come in on a single rod scram.
Plausible: 2 out of 3 are correct. SCRAM AIR HEADER LOW PRESSURE comes in on a full scram.

C. **Correct:** CRD ACCUM TROUBLE will come in due to low accumulator pressure following the rod scram. ROD DRIFT will come in since the rod is moving without an RMCS command. ROD OUT MOTION BLOCK comes from the RWM and/or SDV not drained.

D. **Incorrect**: ROD OVERTRAVEL is indicative of an uncoupled control rod NOT a scrammed rod.

Plausible: 2 out of 3 are correct. Rods travel slightly beyond 00 on a scram and settle at 00 once the scram is reset, so ROD OVERTRAVEL is a plausible misconception.

Level: RO Exam

Lesson Plan Objective: ABIC01E001
Recognize abnormal indications/alarms and/or procedural requirements for implementing Control Rod.

Source: Mod HC Bank Q55937

Level of knowledge: Analysis

Reference(s): HC.OP-AB-IC-0001 rev. 4 - Alarms

KA: G.2.4.46
2.4.46 Ability to verify that the alarms are consistent with the plant conditions. (CFR: 43.5 / 45.3 / 45.12) IMPORTANCE RO 3.5 SRO 3.6

Comments / Change Record:

- None

Question: 76 Answer: B

1 Pt(s) Given the following:

- The Unit is in OPCON 1 at 100% power.
- Multiple CRIDS high temperature alarms are in for both the "A" and "B" Reactor Recirculation Pumps, and the affected temperature indications are trending up on the CRIDS display.
- The following points are trending up on both pumps:
  - Pump Motor Oil high temperature
  - Upper & Lower Motor bearing high temperature
  - # 1 & #2 Seal Cavity high temperature
- The plant is currently stable.

Which one of the following malfunctions could explain this combination of indications, and under theses circumstances, what is the MAXIMUM amount of time that plant procedures permit continued Recirculation Pump operations?

A. **Inadvertent isolation of Chilled Water flow to the Recirculation Pumps. Restore cooling within 10 minutes, or trip both Recirculation Pumps.**

B. **Inadvertent isolation of RACS cooling water flow to the Recirculation Pumps. Restore cooling within 10 minutes, or trip both Recirculation Pumps.**

C. **Inadvertent isolation of Chilled Water flow to the Recirculation Pumps. Restore cooling within 20 minutes, or trip both Recirculation Pumps.**

D. **Inadvertent isolation of RACS cooling water flow to the Recirculation Pumps. Restore cooling within 20 minutes, or trip both Recirculation Pumps.**

Distracter Analysis:

A. **Incorrect**: Chilled Water normally supplies cooling to the Recirc Pump Motor AIR Coolers. A loss of Chilled Water would result in Motor WINDING high temperature indications and alarms NOT the indications presented in this scenario.
Plausible: Chilled water cools the motor air coolers, and it is often confused with RACS. RACS can provide backup cooling to drywell

chilled water loads and this contributes to the misconception. In addition, the maximum time specified is correct for a loss of RACS.

B. **Correct:** RACS cooling supplies cooling water to the specified loads. HC.OP-AB.COOL-003, "Reactor Auxiliary Cooling" requires that the Recirc Pumps be tripped if RACS cooling CANNOT be restored within 10 minutes.

C. **Incorrect**: Chilled Water normally supplies cooling to the Recirc Pump Motor AIR Coolers. A loss of Chilled Water would result in Motor WINDING high temperature indications and alarms. In addition, the maximum time specified is incorrect.
Plausible: Chilled water cools the motor air coolers, and it is often confused with RACS. RACS can provide backup cooling to drywell chilled water loads and this contributes to the misconception.

D. **Incorrect**: The maximum time specified is incorrect.
Plausible: RACS is correct.

Level: SRO Exam
CFR 55.43(5)

Lesson Plan Objective: RACS00E019
Explain the effect that a loss of RACS has on the Reactor Recirculation System, including any time restraints that may or may not be placed on operation of the Reactor Recirculation pumps. IAW available control room references

Source: New

Level of knowledge: Comprehension

Reference(s):
Lesson Plan NOH01RECIRC, "Recirculation System"
Lesson Plan NOH01RACS00C, "Reactor Auxiliary Cooling"
Lesson Plan NOH01TBCW00, "Turbine Building Chilled Water"
HC.OP-AB.COOL-003, "Reactor Auxiliary Cooling"

KA: 295018.A2.03
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER: (CFR: 41.10 / 43.5 / 45.13) AA2.03 Cause for partial or complete loss 3.2 3.5

Comments / Change Record:

- None

Question: 77 Answer: A

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1 Pt(s) Technical Specification LCO 3.6.1.7 states that drywell average air temperature shall not exceed 135°F.

The basis for this LCO is.....

A. to ensure that containment peak air temperature does not exceed the design temperature during LOCA conditions.

B. to ensure that containment peak air temperature does not exceed the Tech Spec Drywell Temperature Safety Limit during LOCA conditions.

C. to ensure that the containment can absorb the associated decay and structural sensible heat released during a reactor coolant system LOCA blowdown.

D. to ensure that RPV level instrument reference legs do not flash during a reactor coolant system LOCA blowdown.

Distracter Analysis:

A. Correct: Reference Tech Spec Basis for LCO 3.6.1.7.

B. Incorrect: There is no Tech Spec Safety Limit for drywell temperature.
Plausible: There are Tech Spec Safety Limits for the fuel barrier and RPV barrier, so it would be a plausible misconception for someone to think that there would be a Tech Spec Safety Limit for Drywell pressure and or temperature. In addition, there is a Limiting Safety System Setting (LSSS) for drywell pressure.

C. Incorrect: This is an excerpt from the Depressurization Systems LCO Basis 3.6.2 and it is related to the suppression pool water heat capacity. It may be a partially true statement, but it is not the basis for the drywell air temperature.
Plausible: This is true regarding suppression pool water temperature which is part of containment.

D. Incorrect: Flashing is not an issue at 135°F, and flashing is not discussed in the Tech Spec Bases.
Plausible: Drywell temperatures above saturation can cause flashing, and it is a concern in the EOPs.

Level: SRO Exam
CFR 55.43(2)

Lesson Plan Objective: TECSPCE009
Explain the bases for Hope Creek Generating Station Technical Specification Safety Limits and Limiting Safety System Settings. **(SRO/STA Only)**

Source: New

Level of knowledge: Memory

Reference(s):
Technical Specification Bases

KA: 295028.G2.2.25
2.2.25 Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (CFR: 43.2)

Comments / Change Record:

- None

Question: 78 Answer: C

1 Pt(s) At 18:00, you receive the following information at shift turnover:

- The Unit is in OPCON 4 and was shutdown earlier in the day to repair a known coil leak on Drywell Cooling Unit 1AVH212.
- "A" Loop RHR is in Shutdown Cooling with SDC flow being maintained at 7000 GPM.
- Reactor coolant temperature is 175°F at the "A" RHR HX inlet.
- RPV level is steady at 80 inches on the Shutdown Range.
- RPV pressure is steady at 0 psig on the feedwater CRT digital display.
- Reactor head vents are open.
- Both Reactor Recirculation Pumps are out of service.

Later in the shift:

- At 23:00, the Reactor Operator observed RPV pressure change from 0 psig to 5 psig over the last hour on the feedwater CRT digital display along with a noticeable rise in drywell cooler drain flow.
- Reactor coolant temperature is still steady at 175°F at the "A" RHR HX inlet.

Which one of the following explains the RPV pressure indication and Shutdown Cooling status?

A. **The RPV is NOT steaming and the pressure indication is normal at the low end of instrument range. SDC flow is too low; however, core circulation is satisfactory since reactor coolant temperature is steady and well below 200°F as measured at the RHR HX inlet.**

B. **The RPV is NOT steaming and the pressure indication is normal at the low end of instrument range. SDC indications are normal since reactor coolant temperature is steady and well below 200°F as measured at the RHR HX inlet.**

C. **The RPV is steaming. SDC flow is too low to ensure adequate core circulation and flow must be raised.**

D. **The RPV is steaming. RPV level is too low to ensure adequate core circulation and level must be raised.**

Distracter Analysis:

A. **Incorrect**: Core circulation is NOT satisfactory with SDC flow less than 10,000 GPM, and RHR HX inlet temperature may not be representative of the bulk coolant temperature, so temperature stratification may occur. RPV pressure instruments may not read exactly 0 psig; however, a 3 trend up over an hour may indicate that steaming is occurring.
Plausible: SDC flow is low and RHR HX inlet temperatures suggest adequate core circulation and coolant temperature less than 200°F.

B. **Incorrect**: Core circulation is NOT satisfactory with SDC flow less than 10,000 GPM, and RHR HX inlet temperature may not be representative of the bulk coolant temperature, so temperature stratification may occur. RPV pressure instruments may not read exactly 0 psig; however, a 3 trend up over an hour may indicate that steaming is occurring.
Plausible: RHR HX inlet temperatures suggest adequate core circulation and coolant temperature less that 200°F, so someone could infer that SDC indications are normal.

C. **Correct:** SDC flow is too low to ensure adequate core circulation. The slight rise in RPV pressure along with increasing drywell drain flow indicates that there is temperature stratification in the RPV and that RCS coolant temperature may be above 212°F near the surface resulting in steaming through the RPV vents to the drywell. The RHR HX inlet temperature may not be indicative of bulk coolant temperature at reduced shutdown cooling flow (less than 10.000 GPM).

D. **Incorrect**: Reactor level is maintained at 80 inches to ensure that a natural circulation path exists if forced circulation is lost, so it is at the correct value.
Plausible: The slight rise in RPV pressure is an indication that steaming may be occurring in the RPV. In addition, level is a parameter important to natural circulation.

Level: SRO Exam
CFR 55.43(2) and CFR 55.43(5)

Lesson Plan Objective: RHRSYSE009
Given plant problems/industry events associated with the Residual Heat Removal System:

a. Discuss the root cause of the plant problem/industry event IAW the associated plant problems/industry event document.
b. Discuss the HCGS design and/or procedural guidelines that mitigate/reduce the likelihood of the problem/industry event at HCGS IAW the associated plant problems/industry event document.
c. Discuss the "lessons learned" from this problem/event IAW the associated plant problems/industry event document.

Source: New

Level of knowledge: Analysis

Reference(s):
HC.OP-SO.BC-0002, "Decay Heat Removal Operations"
Lesson Plan NOH01RHRSYS, "Residual Heat Removal"

KA: 295021.A2.06
Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING : (CFR: 41.10 / 43.5 / 45.13)
AA2.06 Reactor pressure 3.2 3.3

Comments / Change Record:

- Capitalized SD Range, initial pressure lowered to 0 psig, pressure then rose for 0 psig to 5 psig, capitalized NOT in the answers. (licensee comments)

Question: 79 Answer: C

1 Pt(s) Given the following:

- The Unit is in OPCON 1 at 100% power.
- A loss of all drywell cooling occurs.
- Drywell pressure rises to 1.4 psig and operators manually scram the reactor.
- RPV level dropped to a low of -10 inches and was restored to +30 inches with feedwater.
- Drywell pressure continues to rise to 2.0 psig due solely to the loss of drywell cooling.
- All automatic actions occur as designed.
- HPCI injected and was immediately secured since level control was established with feedwater.
- Drywell cooling was restored several minutes later, drywell pressure was reduced, all isolations were reset, and the plant was stabilized in OPCON 3.

Which one of the following is the HIGHEST level event classification that applies the event described above?

A. 15 minute EMERGENCY notification to the states & counties.

B. 1 hour NON-emergency notification to the NRC Operations Center.

C. 4 hour NON-emergency notification to the NRC Operations Center.

D. 8 hour NON-emergency notification to the NRC Operations Center.

Distracter Analysis:

A. **Incorrect**: This event does not meet an EAL threshold; therefore, no EMERGENCY notifications are required. The high drywell pressure condition resulted from a loss of drywell cooling and was NOT indicative of a loss or potential loss of the RCS barrier. See the basis for EAL # 3.2.2.b.
Plausible: There is an EAL for high drywell pressure due to a loss of the RCS barrier. If an actual RCS leak was the cause of the condition, then ALERT declaration would be required and that

would require 15 minute emergency notification to the states and counties.

B. **Incorrect**: Does not meet the threshold for 1 hour notification.
Plausible: 1 hour non-emergency notifications do exist, and in the past, ECCS injection to the vessel did require 1 hour notification.

C. **Correct:** 4 hour NON-emergency notification to the NRC Operations Center due to the valid HPCI actuation and injection to the vessel and due to the manual RPS actuation in anticipation of the automatic scram on 1.68 psig. The HPCI initiation was valid since the signal was generated by the sensor which measured a physical system parameter that was at its set point. See RAL #11.3.1 and 11.3.2.

D. **Incorrect**: The 4 hour report is the most limiting reportability requirement in this case.
Plausible: The primary containment isolation on 1.68 psig qualifies for an 8 hour notification.

Level: SRO Exam
CFR 45.3

Lesson Plan Objective: NOH04ADA120C-03
From memory, describe the actions for Operations Shift Management Review of an issue to determine the following:

- If operability is required
- Determine Operability
- Reportability requirements

IAW LS-AA-120.

Source: New

Level of knowledge: Analysis

Reference(s):
Hope Creek Event Classification Guide & Bases

KA: 295024.G2.4.30
2.4.30 Knowledge of which events related to system operations/status should be reported to outside agencies. (CFR: 43.5 / 45.11)

Comments / Change Record:

- Added words in the stem to be clear that the rise in drywell pressure was "due solely to the loss of drywell cooling". (licensee comment 08/09)
- Change "MOST LIMITING reportability requirement" to "HIGHEST level event classification". (licensee comment 08/09)

Question: 80 Answer: D

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1 Pt(s) Given the following conditions:

- The plant is operating at 100% power.
- All systems are aligned for normal operations.
- A fire started under Control Room console 10C651 causing a reactor scram.
- The Control Room has been evacuated because of extreme smoke conditions.
- The reactor has been depressurized to less than 80 psig with SRV's and RCIC.
- "B" RHR is in Suppression Pool Cooling from the RSP.

Which one of the following describes the preferred Remote Shutdown system to be used to achieve Cold Shutdown, and what is the maximum cooldown rate that is permitted in accordance with HC.OP-IO.ZZ-0008, "Shutdown from Outside the Control Room"?

A. **"A" RHR in Shutdown Cooling at less than 100 °F/hr.**

B. **"A" RHR in Shutdown Cooling at less than 90 °F/hr.**

C. **"B" RHR in Shutdown Cooling at less than 100 °F/hr.**

D. **"B" RHR in Shutdown Cooling at less than 90 °F/hr.**

Distracter Analysis:

A. **Incorrect**: "B" SDC is the preferred SDC loop to be used if at all possible as directed by the procedure. The procedure directs less than 90 °F/hr.
Plausible: Provisions are provided in the procedure to us "A" SDC, but the procedural actions are more complex and time consuming than the "B" loop. The Tech Spec limit is less than 100 °F/hr.

B. **Incorrect**: "B" SDC is the preferred SDC loop to be used if at all possible as directed by the procedure.
Plausible: Provisions are provided in the procedure to us "A" SDC, but the procedural actions are more complex and time consuming than the "B" loop. The rate is correct.

C. **Incorrect**: The procedure directs less than 90 °F/hr.
Plausible: "B" SDC is correct, and the Tech Spec limit is less than 100 °F/hr.

D. **Correct:** HC.OP-IO.ZZ-0008, "Shutdown from Outside the Control Room" step 5.9.6 specifies that "B" RHR be secured from Suppression Pool Cooling and be placed in Shutdown Cooling mode. The procedure directs that the cooldown be maintained at less than 90 °F/hr.

Level: SRO Exam
CFR 55.43(b)(5)

Lesson Plan Objective: REMS/DE002
Given a system that is controlled from, or is required to support the operation of the Remote Shutdown System, explain the function of the supporting system.
IAW available control room references.

Source: Mod Hope Creek Bank (Q56470)
Audit Exam 9/99

Level of knowledge: Memory

Reference(s):
NOH01REMS/D, "Remote Shutdown"
HC.OP-IO.ZZ-0008, "Shutdown from Outside the Control Room"

KA: 295016.A2.06
Ability to determine and/or interpret the following as they apply to CONTROL ROOM ABANDONMENT : (CFR: 41.10 / 43.5 / 45.13) AA2.06 Cooldown rate 3.3 3.5

Comments / Change Record:

- Licensee commented that this was a difficult memory level question. Comment noted, but no changes made. This question was developed / modified form the Hope Creek bank. 90 °F is a standard admin cooldown rate used at HC and it is used in EOPs and other procedures, so this should be able to be recalled based on extensive repetitive training. In addition, the candidate should have knowledge of the basic strategy and preferred or primary systems used while executing remote shutdown from lesson plans, etc. without having to have the procedure "memorized". (licensee comment 09/11)

Question: 81 Answer: B

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1 Pt(s)

During a plant startup at 5% reactor power, the on-shift Chemistry Technician reports the following results with regards to the Standby Liquid Control Storage Tank:

- NET TANK VOLUME - 4700 gallons
- % WEIGHT SOLUTION CONCENTRATION - 13.7%
- SLC Tank temperature is 80°F

Which of the following describes the SLC system status and the basis for that conclusion?

A. **The SLC system is operable since there is sufficient boron to bring the reactor from 5% power to a cold, Xenon free shutdown assuming that withdrawn control rods remain fixed in the 5% power pattern.**

B. **The SLC system is NOT operable since there is insufficient boron to bring the reactor from full power to a cold, Xenon free shutdown assuming that withdrawn control rods remain fixed in the rated power pattern.**

C. **The SLC system is NOT operable since the solution is OVER saturated, so boron may precipitate out of solution such that there will NOT be enough boron to overcome the reactivity insertion rate due to cooldown and Xenon decay.**

D. **The SLC system is operable since the solution is UNDER saturated, so boron will NOT precipitate out of solution so there will be enough boron to overcome the reactivity insertion rate due to cooldown and Xenon decay.**

Distracter Analysis:

A. **Incorrect**: The Reactor is in OPCON 1, so SLC must meet the Tech Spec operability requirements regardless of the low power level. **Plausible**: In reality, there is probably enough SLC to shutdown the reactor from 5% initial power.

B. **Correct:** The SLC volume is too low for the existing concentration, so there is insufficient boron to ensure that the reactor can be brought from full power to a cold, Xenon free shutdown assuming that withdrawn control rods remain fixed in the rated power pattern.

C. **Incorrect**: The system is inoperable based on the total amount of boron currently available to inject. Precipitation should not be a problem with tank temperature at 80°F.
Plausible: The system is inoperable. Precipitation can be a problem at less than 70°F. In addition, some of the words regarding Xenon and reactivity insertion rate were pulled from the Tech Spec Bases.

D. **Incorrect**: The SLC system is NOT operable.
Plausible: An under saturated solution would be less susceptible to precipitation, so if someone thought precipitation was the issue, and if they concluded the solution was under saturated, then they could conclude the system was operable. In addition, some of the words regarding Xenon and reactivity insertion rate were pulled from the Tech Spec Bases.

Level: SRO Exam
CFR 55.43(b)(2)

Lesson Plan Objective: SLCSYSE018
Provided access to Technical Specifications, analyze conditions to determine if SLC System parameters are within the Acceptable Operating Region of the Sodium Pentaborate Solution Volume/ Concentration Requirements Graph, IAW HCGS Technical Specifications.

Source: Mod HC Bank Question Q56443

Level of knowledge: Analysis

Reference(s):
Tech Specs & bases

KA: 295037.G2.1.33
2.1.33 Ability to recognize indications for system operating parameters which are entry-level conditions for technical specifications. (CFR: 43.2 / 43.3 / 45.3) SRO 4.0

Comments / Change Record

- Capitalized NOT in distracter C. (licensee comment 08/09)

Question: 82 Answer: B

1 Pt(s) Given the following:

- The Unit was operating at 80% power.
- A Main Steam line rupture occurred in the turbine building.
- A Group I Primary Containment Isolation signal was received on High Main Steam Line Flow.
- The MSIV's failed to completely isolate the leak.
- The reactor automatically scrammed.
- Reactor level dropped to -45 inches, and level was restored to +30 inches with HPCI & RCIC.
- Primary & Secondary containment parameters remained normal.
- Personnel reported a large steam plume in the turbine building upstream of the turbine stop valves
- The reactor has been depressurizing through the steam line rupture for the last 75 minutes.

Current conditions (75 minutes after the event began):

- Reactor pressure is 200 psig and dropping slowly.
- The steam line leak is still in progress.
- Reactor level is +30 inches with condensate injecting for level control.
- HPCI & RCIC have been secured to slow the depressurization rate, and both systems were realigned for automatic operations.
- Primary & Secondary containment parameters are normal.
- Radiation Protection has just provided a Dose Assessment projection based on Plant effluent sample analysis as follows:
  - o 3.4E-01 mRem TEDE 4-day dose at the MEA
  - o 1.6E-01 mRem Thyroid-CDE

Which one of the following describes the HIGHEST event classification and the EOP(s) that were or are required to be entered to mitigate the event?

A. General Emergency. EO.ZZ-0101 & EO.ZZ-0103/4.

B. Site Area Emergency. EO.ZZ-0101 only.

C. Alert. EO.ZZ-0101 only.

D. Unusual Event. EO.ZZ-0101 & EO.ZZ-0103/4.

Distracter Analysis:

A. **Incorrect**: The threshold was not reached for GE classification, and EOP-104 entry was not required. See justification for B.

B. **Correct:** SAE declaration required due to a Main Steam Line break outside containment with a failure of the MSIV's to isolate leak with a downstream pathway to the environment. EOP-0101 was required to be entered on low reactor level; however, EOP-0104 entry was/is NOT required since the dose projection has only reached the Unusual Event level based on TEDE. EOP-0104 entry is required when there is a gaseous radioactive release at the ALERT threshold. The SAE declaration was based on the loss of the RCS and containment barriers NOT the radioactive release, so EOP-104 entry was/is not required.

C. **Incorrect**: SAE threshold was reached. See justification for answer B.

D. **Incorrect**: SAE threshold was reached & EOP-104 entry was not required. See justification for answer B.
Plausible: UE is correct just for the rad release, but this was not the highest EAL for the event.

Level: SRO Exam
CFR 55.43(b)(5)

Lesson Plan Objective: ABCNT4E001
Recognize abnormal indications/alarms and/or procedural requirements for implementing Radioactive Gaseous Release.

Source: New

Level of knowledge: Analysis

Reference(s):
Hope Creek Event Classification Guide
HC.OP-EO.ZZ-0101 & 0103/4.

KA: 295038.A2.03
Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE: (CFR: 41.10 / 43.5 / 45.13)
AA2.03 †Radiation levels: Plant-Specific 3.1 3.9

Comments / Change Record:

- Changed “Health Physics” to “Radiation Protection”
- Changed procedure # from EO.ZZ-0104 to EO.ZZ-0103/4. (licensee comment 08/09)

Question: 83 Answer: D

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1 Pt(s) While performing a panel walk down at 100% power you observe the following:

- "A" SPTMS Avg Suppression Pool temperature is 82°F.
- "B" SPTMS Avg Suppression Pool temperature is 85°F.
- After checking the daily control console logs, you identify that the "B" SPTMS Avg has risen 3°F over the last 24 hours.
- You then check the ***individual*** SPTMS channel inputs and identify that SPTMS channel B6 is reading 100°F while all of the other ***individual*** SPTMS channel inputs show indications that range between 80°F and 85°F.
- SPTMS channel B6 is located near Torus Azimuth 0° which is in close proximity to SRV tailpipe F013H.
- SRV F013H tailpipe temperature is currently reading 180°F and it has risen 40°F over the last 24 hours as recorded in the daily console logs.
- All other plant indications are normal.

Which one of the following is the correct action to take in response to these indications?

A. **Enter HC.OP-AB.RPV-0006, "Safety / Relief Valve" AND execute the immediate operator actions.**

B. **Declare "B" SPTMS inoperable AND enter the LCO for having less than the required number of operable Suppression Chamber Water Temperature channels.**

C. **Enter HC.OP-EO.ZZ-0102, "Primary Containment Control" due to high local Suppression Pool Water Temperature AND place Suppression Pool Cooling in service to lower the pool temperature.**

D. **Notify the SRV System Manager AND continue to monitor Suppression Pool Temperature AND F013H SRV tailpipe temperature.**

Distracter Analysis:

A. **Incorrect**: This procedure is for an open SRV. There is indication of SRV leakage, but there are no other indications to conclude that the SRV is open (no acoustics alarm, no change in steam flow/feed flow, no change in pressure, etc.). Immediate actions would require reducing power and cycling the SRV control switch which would cause the SRV to open and possibly not reseat.
Plausible: The temperature indications would be used to confirm that an SRV was open if there were other parameters or alarms indicating that an SRV opened.

B. **Incorrect**: The SPTMS average temperature and the individual channel input are reflecting actual plant conditions, so they are NOT inoperable.
Plausible: Tech Specs require 2 "channels" of SPTMS indication for average bulk temperature. In addition, someone could conclude that having one individual channel input reading high relative to the other inputs would render that channel inoperable.

C. **Incorrect**: The EOP entry is based on average Suppression Pool temperature greater than 95°F. In this case, average temperature on the "B" SPTMS is reading 85°F, so there is no emergency or reason to enter the EOP. The operators may elect to start Suppression Pool Cooling to lower the local temperature, but that can be done using normal plant procedures rather than Emergency Operating Procedures.
Plausible: Local temperature is actually high, and starting suppression pool cooling would lower the temperature.

D. **Correct:** The indications suggest that the SRV has pilot or main seat leakage. The daily logs have a note that the System Manager should be contacted if there is a greater than 30°F rise over the normal or baseline temperature. A reference is not needed to determine the correct answer. The symptoms alone indicate that the local SP temperature is rising due to SRV leakage near the SP temperature detector. The operator would need several other diverse indications to conclude that the SRV was open, and none of those indications exist at this time. Notifying the System Manager & increasing monitoring is the prudent and correct action to take based on the indications presented.

Level: SRO Exam
CFR 55.43(b)(2) & (5)

Lesson Plan Objective:

- EOP102E004: From memory, recall the reason why average suppression pool temperature is used for determining the entry condition and subsequent actions IAW the Primary Containment Control - Suppression Pool Lesson Plan.
- ABRPV6E001: Recognize abnormal indications/alarms and/or procedural requirements for implementing Safety/Relief Valve.

Source: New

Level of knowledge: Analysis

Reference(s):
HC.OP-AB.RPV-0006, "Safety / Relief Valve"
HC.OP-EO.ZZ-0102, "Primary Containment Control"
HC.OP-DL.ZZ-0003, "Log 3 Control Console Log Condition 1, 2, & 3"

KA: 295013.A2.02
Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE: (CFR: 41.10 / 43.5 / 45.13) AA2.02 Localized heating/stratification.3.2 3.5

Comments / Change Record:

- Capitalized "AND" in answer D. (licensee comment 08/09)

Question: 84 Answer: A

1 Pt(s) Given the following conditions:

- A LOCA has resulted from a seismic event
- Reactor water level is -20 inches and rising
- Reactor pressure is 850 psig and slowly lowering
- Drywell Pressure is 31 psig and slowly rising
- Drywell temp is 275 °F and slowly rising
- Suppression Chamber pressure is 30 psig and slowly rising
- Suppression Pool water level is 77 inches
- "B" RHR Loop is in Drywell Spray
- The Main Condenser is NOT available
- All control rods are full in

Based on the above conditions, when is depressurization of the reactor required and with what method?

A. Immediately using 5 ADS Valves

B. Immediately using Turbine Bypass Valves

C. When Drywell Press reaches 35 psig using Turbine Bypass Valves

D. When Drywell Temp reaches 310F using 5 ADS Valves

Distracter Analysis:

A. **Correct:** Immediately using 5 ADS Valves. Emergency de-pressurization must occur now for exceeding the PSP curve.

B. **Incorrect**: Immediately using Turbine Bypass Valves. Emergency de-pressurization can no longer be anticipated. Emergency de-pressurization must occur now for exceeding the PSP curve.

C. **Incorrect**: When Drywell Press reaches 35 psig using Turbine Bypass Valves. Emergency de-pressurization can no longer be anticipated. Emergency de-pressurization must occur now for exceeding the PSP curve.

D. **Incorrect**: When Drywell Temp reaches 310°F using 5 ADS Valves. Emergency de-pressurization must occur now for exceeding the PSP curve. Emergency de-pressurization for high Drywell temperature does not occur until 340°F.

Level: SRO Exam
CFR 55.43(b)(5)

Lesson Plan Objective: EO102PE007
Given any step of the procedure, determine the reason for performance of that step and/or predict expected system response to control manipulations prescribed by that step IAW the Primary Containment Control - Drywell Lesson Plan.

Source: HC Bank (Q56157)
Audit Exam 1999

Level of knowledge: Comprehension

Reference(s):
HC.OP-EO.ZZ-0102(Q) curve SCP-L

KA: 295010.G2.4.6 Knowledge symptom based EOP mitigation strategies. (CFR: 41.10 / 43.5 / 45.13)

Comments / Change Record:

- None

Question: 85 Answer: A

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1 Pt(s) Given the following conditions:

- A large break LOCA has occurred inside the Drywell.
- Multiple equipment failures occurred.
- Drywell pressure is 15 psig.
- Steam cooling was required until water level was restored above TAF with Fire Water.
- The Containment H2/O2 Analyzers were placed in-service 2 hours ago.
- The containment H2/O2 Analyzers have just alarmed on High Drywell H2 Concentration and the trend is upward at 0.5 percent per hour.
- H2 concentration is at 2.1% and O2 concentration is 1.3%.
- The H2 Recombiners are currently NOT in service.

Which one of the following actions is required IAW HC.OP-EO.ZZ.0102?

A. **Exit EOP-102 and enter SAG since core damage is in excess of what the Emergency Operating Procedures were typically designed to handle.**

B. **Place the H2 Recombiners in service since there are sufficient quantities of H2 & O2 to support effective recombination.**

C. **Exit EOP-102 and enter SAG since the lower H2 detonation limit has been reached and containment failure may occur with known fuel damage.**

D. **Vent containment via the Suppression Chamber to reduce Drywell H2 concentration since the Recombiners could initiate an H2 detonation at these concentrations.**

Distracter Analysis:

A. **Correct:** EOP-102 PC/H-1 directs that the EOP be exited and SAG entered is H2 concentration exceeds 2%. In addition, the bases states that a 2% H2 concentration confirms fuel damage above 10 CFR 50.46 ECCS design requirements and beyond what the EOPs were typically designed to handle.

B. **Incorrect**: With concentrations above 2%, the direction is to exit the EOP and transition to SAG. The Recombiners could have potentially been placed in service earlier in accordance with EOP-

102, but the stem indicates that the recombiners are currently out of service. Now that H2 concentration is above 2%, EOP-102 must be exited. Recombiner operations will be governed by SAG at this point NOT EOP-102.
Plausible: EOP-102 does direct that the Recombiners be placed in service if concentration is above 0.5% since H2 is present in sufficient quantities for the Recombiners to be effective.

C. **Incorrect**: Correct action but wrong reason. H2 and O2 concentrations are well below the LEL for H2.
Plausible: Action is correct. In addition, containment failure would be possible if there were a detonation.

D. **Incorrect**: Venting is not an option in EOP-102 under the current conditions.
Plausible: Sag may direct venting to control H2 concentration at some point. In this event, then Suppression Chamber venting would be a good idea to allow scrubbing in the pool and the humid environment and sprays could be used as a means to prevent or mitigate am H2 burn.

Level: SRO Exam
CFR 55.43(b)(5)

Lesson Plan Objective: EO102PE004
Recall the reasons why the following are used for determining the entry condition and / or subsequent actions IAW the Primary Containment Control - Drywell Lesson Plan.
a. Drywell Pressure
b. Average Drywell Temperature
c. H2 and O2 concentrations in the drywell

Source: Mod HC Bank Question (Q68880)
NRC exam 2002

Level of knowledge: Comprehension

Reference(s):
HC.OP-EO.ZZ.0102 & Bases

KA: 500000.A2.03
Ability to determine and/or interpret the following as they apply to HIGH CONTAINMENT HYDROGEN CONCENTRATIONS:
(CFR: 41.10 / 43.5 / 45.13) EA2.03 Combustible limits for drywell
3.3 3.8

Comments / Change Record:

- None

Question: 86 Answer: C

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1 Pt(s) Given the following:

- The reactor is at 27% power.
- Power ascension is in progress following a major refueling outage.
- Main Turbine Generator bearing number 10 and 11 were replaced during the outage.
- TURBINE GENERATOR VIB HI alarm **D3-C5** just annunciated.
- Number 10 bearing is at 10 mils and rising slowly.
- Number 11 bearing is at 9 mils and rising slowly.

Which one of the following describes the required immediate actions?

A. **REDUCE Recirculation Pump speed to MINIMUM, then LOCK the Mode Switch in Shutdown, AND then TRIP the Main Turbine.**

B. **IMMEDIATELY TRIP the Main Turbine, AND then LOCK the Mode Switch in Shutdown.**

C. **LOCK the Mode Switch in Shutdown, AND then TRIP the Main Turbine.**

D. **REDUCE reactor power to less than 25%, AND then TRIP the MAIN Turbine.**

Distracter Analysis:

A. **Incorrect**: This is NOT the *immediate action* for when vibration is greater than or equal to *8 mils* on Number 11 or 12 bearing.
Plausible: This is the *retainment override* for the condition when vibration is greater than or equal to 7 mils on Number 11 or Number 12 bearing; however, vibration is now at 8 mils.

B. **Incorrect**: Wrong sequence. Would result in reactor high pressure and high neutron flux on the turbine trip.
Plausible: Could confuse the sequence since there is a problem on the turbine requiring immediate action.

C. **Correct:** With Number 11 bearing above 8 mils with power greater than 25%, the immediate action is to lock the mode switch in shutdown, then trip the turbine.

D. **Incorrect**: Procedure does NOT permit a controlled power reduction with the given vibration levels.

Plausible: It would be nice to do this since the turbine could be tripped while leaving the reactor online on bypass valves.

Level: SRO Exam

Lesson Plan Objective: ABBOP2E003
From memory, recall the Immediate Operator Actions for Main Turbine.

Source: New

Level of knowledge: Memory

Reference(s):
HC.OP-AB.BOP-0002, "Main Turbine"

KA: 245000.A2.09
Ability to (a) predict the impacts of the following on the MAIN TURBINE GENERATOR AND AUXILIARY SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
(CFR: 41.5 / 45.6) A2.09 Turbine Vibration 2.5 2.8

Comments / Change Record:

- Capitalized the word "AND" in all answers. (licensee comments)
- Deleted "and" in the second clause of answer "A". (licensee comments 08/28)

Question: 87 Answer: B

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1 Pt(s) Given the following:

- A reactor startup from COLD SHUTDOWN is in progress.
- IRM B is inoperable and was bypassed prior to startup.
- The point of adding heat has just been reached.

Then,

- IRM H is observed by the operators to fail downscale, producing a DOWNSCALE alarm but no INOP trip.

As far as Technical Specifications are concerned, the reactor startup:

A. is NOT permitted to continue. Only rod motion by scram is permitted.

B. may continue if IRM H is placed on Range 1.

C. may continue but the mode switch cannot be placed in the RUN position until one of the inoperable IRMs is returned to service.

D. is NOT permitted to continue. The control rods may be inserted in the reverse sequence to shutdown the reactor.

Distracter Analysis:

A. **Incorrect**: Continued operation is permitted. See B. Also, normal rod motion is permitted.

B. **Correct:** Continued startup is permitted. **Regarding Rod Block Instrumentation:** Placing on Range 1 bypasses the downscale rod block to allow or ensure continued rod withdrawal. Only required to have 6 operable channels per trip function, and 6 remain. **Regarding RPS instrumentation:** Tech Spec 3.3.1.a and table 3.3.1-1. Table **3.3.1-1** states for Op Con 2 need 3 IRMs per trip system. **3.3.1.a** states with the number of operable channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel (s) and/or that trip system in the tripped condition* within twelve hours. The provisions of Specification 3.0.4 are **not applicable**. **3.0.4** Entry into an OPERATIONAL CONDITION or other specified condition shall **not** be made when the conditions for the Limiting Condition for Operation are not met and the associated

ACTION requires a shutdown if they are not met within a specified time interval.

C. **Incorrect**: Can place the mode switch in RUN since the IRM's are automatically bypassed with the mode switch in RUN.

D. **Incorrect**: Continued operation is permitted. See B.

Level: SRO Exam
CFR 55.43(b)(2)

Lesson Plan Objective: IRMSYSE006
Given a scenario of applicable operating conditions:

a. Choose those sections which are applicable to the IRM System
b. Evaluate IRM operability and determine required actions based upon system operability
c. Determine the bases for those tech spec items associated with the IRM System IAW HCGS Technical Specifications. (SRO Only)

Source: HC Bank (Q54887)

Level of knowledge: Analysis

Reference(s):
NOH05000014C - Intermediate Range Monitoring System

KA: 215003.G2.1.12
Ability to apply Technical Specifications for a system. (CFR: 43.2 / 43.5 / 45.3)

Comments / Change Record:

- Deleted words from Tech Specs in answer B for psychometric balance since they are not need to answer the question. (NRC reviewer comments 08/09)
- A licensee validator commented that there is no specific procedural direction to perform answer B. Modified the stem to state, "As far as Technical Specifications are concerned, the reactor startup:" . In addition, the words "by Technical Specifications" was replaced by "to continue" to eliminate redundant wording. This fix should clarify that this question is based on actions that could be taken as far as Tech Specs are concerned. Procedures could be changed as needed so that the startup can continue since the actions are NOT prohibited by Tech Specs. (licensee comment 09/11)

Question: 88 Answer: A

1 Pt(s) Given the following:

- The plant is operating at 75% power.
- Confirmed seal failures have occurred on the "B" Recirculation Pump.
- The pump has just been tripped.

Which of the following describes the order for isolation of the "B" Recirculation Pump IAW HC.OP-AB.RPV.0003, "Recirculation System / Power Oscillations" and what is the basis for this order?

A. Close the suction valve, then isolate seal purge, and then close the discharge valve. This ensures that that the suction valve is stroked against a minimal differential pressure.

B. Close the suction valve, then isolate seal purge, and then close the discharge valve. This will minimize additional overpressure damage to the seal package.

C. Close the discharge valve, then isolate seal purge, and then close the suction valve. This ensures that that the suction valve is stroked against a minimal differential pressure.

D. Close the discharge valve, then isolate seal purge, and then close the suction valve. This will minimize additional overpressure damage to the seal package.

Distracter Analysis:

A. **Correct:** Correct order and basis. See AB.RPV-0003 and SO-BB-0002 Precaution 3.1.14. The suction valve is not designed to stroke against a high DP and may not fully close against high DP.
B. **Incorrect**: Incorrect basis.
C. **Incorrect**: Incorrect order. Right basis.
D. **Incorrect**: Incorrect order and basis.

Level: SRO Exam
CFR 55.43(b)(5)

Lesson Plan Objective:

- ABRPV3E001: Recognize abnormal indications/alarms and/or procedural requirements for implementing Recirculation System/Power Oscillations.
- RECIRCE013: Given procedure HC.OP-SO.BB-0002, Reactor Recirculation System Operation, explain the bases for listed precautions and limitations.

Source: INPO Bank (HC question 2/28/98)

Level of knowledge: Comprehension

Reference(s):
HC.OP-AB.RPV-0003
HC.OP-SO.BB-0002 Precaution 3.1.4

KA: 202001.A2.10
Ability to (a) predict the impacts of the following on the RECIRCULATION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.10 Recirculation Pump Seal Failure 3.5 3.9

Comments / Change Record:

- Changed the word "discharge" to "suction". (NRC reviewer comment 08/09)
- Reclassified cognitive level from memory to comprehension since the question tests an understanding of the reason behind the sequence of operation. (NRC Exam Author 09/14)

Question: 89 Answer: C

1 Pt(s) Given the following:

- The plant is shutdown for refueling.
- All SRM's are operable.
- The SRM shorting links are installed.
- Arc welding is in progress near the SRM preamp panels.
- An irradiated bundle is being lowered into the core in "C" quadrant.
- The SRM UPSCALE OR INOPERATIVE C3-C1 alarm annunciates.
- "A" SRM indicates hard upscale.
- There was NO change on the other SRMs.

What is the required immediate action?

A. **Raise the fuel bundle up and out of the core and move it to a safe location in the spent fuel storage pool.**

B. **Lower the fuel bundle into the target location AND terminate fuel movement in "A" core quadrant ONLY.**

C. **Terminate fuel movement in all core quadrants AND leave the fuel bundle suspended in place until the cause of the alarm is resolved.**

D. **Terminate welding activities near the SRM preamp panels to stop the spurious alarms. Fuel movement may continue in all quadrants until another alarm occurs.**

Distracter Analysis:

A. **Incorrect**: Not an immediate action.
Plausible: If lowering the bundle caused the upscale, then removing it might correct the problem.

B. **Incorrect**: AB.IC-0004 requires that all fuel movement be terminated.
Plausible: "A" SRM is the only upscale SRM. Other SRMs are normal.

C. **Correct**: Immediate action IAW AB.IC-0004.

D. **Incorrect**: Follow-up action.
Plausible: Follow-up action.

Level: SRO Exam
CFR 55.43(b)(5)

Lesson Plan Objective: IABIC04E003
From memory, recall the Immediate Operator Actions for Neutron Monitoring.

Source: Mod INPO Bank (HC Question 2/23/98)

Level of knowledge: Memory

Reference(s):
HC.OP-AB.IC-0004

KA: 215004.G2.4.49
2.4.49 Ability to perform without reference to procedures those actions that require immediate operation of system components and controls. (CFR: 41.10 / 43.2 / 45.6)

Comments / Change Record:

- Added some additional words to answers B, C, & D for psychometric balance. (NRC reviewer comment 08/09)
- Capitalized "NO" in the last bullet of the stem. (licensee comment 09/11)

Question: 90 Answer: D

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1 Pt(s) Given the following conditions:

- Due to a large reactor coolant leak, suppression chamber pressure was rapidly increasing
- Drywell sprays have been initiated
- Drywell pressure is 10 psig and lowering
- Drywell temperature is 310 °F and lowering

If drywell pressure and temperature lowering results in entering the UNSAFE region of the Drywell Spray Initiation Limit (DWT-P), what action is required?

A. Secure all drywell sprays when the Drywell Spray Initiation Limit curve is reached.

B. Secure drywell sprays at 9.5 psig drywell pressure and lowering.

C. Emergency Depressurize the reactor.

D. Continue drywell sprays until drywell pressure approaches 0 psig.

Distracter Analysis:

A. **Incorrect**: Curve is based on spray initiation, not securing spray.
B. **Incorrect**: DW sprays remain in service until 0 psig. There is no indication in the stem that the RHR pump is needed for adequate core cooling.
C. **Incorrect**: Curve is based on spray initiation, not whether to ED or not. DW temperatures are improving with sprays in service.
D. **Correct:** Once initiated, DW sprays need only be secured BEFORE DW pressure reaches 0 psig. EOP-102, PCC-1, Retainment Override.

Level: SRO Exam
CFR 55.43(b)(5)

Lesson Plan Objective: EO102PE006
Given plant conditions and access to the following curves determine the region of acceptable operation and explain the bases for the curve IAW the Primary Containment Control - Drywell Lesson Plan:
a. Drywell Spray Initiation Limit
b. Pressure Suppression Pressure

Source: HC Bank (Q56794)

Level of knowledge: Comprehension

Reference(s):
EOP-102, PCC-1, Retainment Override.

KA: 226001.A2.15
Ability to (a) predict the impacts of the following on the RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
(CFR: 41.5 / 45.6) A2.15 High Containment / Drywell pressure 3.6 3.8

Comments / Change Record:

- Changed cognitive level from memory to comprehension. The question tests comprehension of the procedural step and the fact that it is permissible to pass through the unsafe region of the DWT-P curve once sprays have been initiated. (NRC Exam Author 09/14)

Question: 91 Answer: C

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1 Pt(s) A Traversing Incore Probe (TIP) trace is in progress when a high drywell pressure event (> 1.68 psig) occurs due to a leak in the recirculation system. Three minutes following the event, the Reactor Operator reports the following indications on the TIP Valve Control Monitor:

- "SQUIB MONITOR" lights - both extinguished
- "SHEAR VALVE MONITOR" lights - both extinguished
- "BALL VALVE OPEN" lights - both illuminated
- "BALL VALVE CLOSED" lights - both extinguished

Which of the following describes the status of the TIP system and the next required operator action(s)?

A. The system has responded as designed. Operator action is required to fire the shear valves.

B. The TIP detectors may NOT have withdrawn. Fire the shear valves, withdraw the remaining cable and then close the ball valves.

C. The TIP detectors may NOT have withdrawn. Withdraw the detectors and verify the ball valves close.

D. The system has responded as designed. Operator action is required to close the ball valves.

Distracter Analysis:

A. **Incorrect**: The system did not respond as required. The ball valve should be closed.
Plausible: Could conclude that squib lights out indicate that the squib fired and the line is isolated.

B. **Incorrect**: The next action is to attempt a manual withdrawal. If that fails, then firing the shear valve alone will isolate the line.
Plausible: Tips may not have withdrawn is correct. The subsequent actions are plausible but not correct.

C. **Correct:** The ball valve should be closed. A failure to auto retract could be the problem. The next action would be to attempt to withdraw the detectors and verify the ball valve closes.

D. **Incorrect**: Did not respond as designed.

Plausible: Could conclude that manual action is needed to close the ball valves. Lights out on shear valve could be interpreted as though the shear fired to isolate the line.

Level: SRO Exam
CFR 55.43(b)(5)

Lesson Plan Objective: TIPS00E003
Given a labeled drawing of the TIP System controls and indications in the main control room (panel 10C607):

a. Explain the function of each indicator.
b. Assess the plant conditions that will cause the indicator to light or extinguish.
c. Predict the effect of each control switch on the TIP System control panels.
d. Select the conditions for permissives required for the control switches to perform their intended function.

Source: HC Bank (Q56564)
NRC Exam 10/99

Level of knowledge: Analysis

Reference(s):
HC.OP-AB.CONT-0002
TIP Lesson Plan

KA: 223002.A2.10
Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.10 Loss of coolant accidents 3.9 4.2

Comments / Change Record:

- Capitalized "NOT" in answers B & C. (licensee comment 08/09)
- Updated reference. (licensee comment 09/11)

Question: 92 Answer: B

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1 Pt(s) The Unit is operating at 100% power when an inadvertent HPCI initiation and injection occurs due to a testing error.

Which one of the following describes the required immediate action?

A. **Reduce feedwater flow as needed to maintain RPV level between Level 4 and Level 7.**

B. **Reduce reactor recirculation flow and insert rods as needed to reduce reactor power.**

C. **Actuate Isolation Logic A and C by depressing the associated trip pushbutton.**

D. **Shut the HPCI pump discharge isolation valve HV-F007 using the Control Room hand switch.**

Distracter Analysis:

A. **Incorrect**: This is an immediate action in AB.RPV-004 for a feedwater control malfunction. Feedwater will automatically control level in this situation.

B. **Correct:** This event results in a cold water injection and positive reactivity insertion. The immediate action per AB.RPV-0001 Reactor Power for a rising power condition is to reduce reactor power as needed below the pre transient value.

C. **Incorrect**: Securing HPCI is a follow-up action not an immediate action. Furthermore, the procedure directs tripping HPCI not isolating it.

D. **Incorrect**: Securing HPCI is a follow-up action not an immediate action. Furthermore, the procedure directs tripping HPCI not isolating the discharge valve.

Level: SRO Exam
CFR 55.43(b)(5)

Lesson Plan Objective: ABRPV1E003
From memory, recall the Immediate Operator Actions for Reactor Power.

Source: INPO Bank (HC Question 12/18/95)

Level of knowledge: Memory

Reference(s):
HC.OP-AB.RPV-0001 Reactor Power

KA: 206000.A2.17
Ability to (a) predict the impacts of the following on the **HIGH PRESSURE COOLANT INJECTION SYSTEM**; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: (CFR: 41.5 / 45.6) A2.17 †HPCI inadvertent initiation: BWR-2,3,4; 3.9 4.3

Comments / Change Record:

- Modified distracter A for psychometric balance. The distracter is now related to level control rather than power oscillations. (NRC reviewer 08/09)

Question: 93 Answer: C

1 Pt(s) The following plant conditions exist:

- Reactor startup is in progress
- Mode switch is in "RUN"
- Reactor power is 30%
- "A" FRVS vent unit was declared inoperable 8 hours ago.
- Repair is expected to be completed in 8 hours.

The "A" EDG capability test was completed at the end of the previous shift. As you review the paperwork you determine that the EDG must be declared inoperable due to unsatisfactory acceptance criteria.

Which ONE of the following actions is required per technical specifications?

A. The reactor must be in Hot Shutdown within 12 hours.

B. The reactor must be in Cold Shutdown within 24 hours.

C. Startup may continue provided the "A" EDG is made operable within the following 72 hours.

D. Suspend power ascension until the "A" EDG and "A" FRVS are declared operable. Operation at the current power level is permitted provided that the "A" EDG is made operable within the following 72 hours.

Distracter Analysis:

A. **Incorrect**: The Unit is not in a shutdown statement. The individual LCOs for the EDG and FRVS apply.
Plausible: Could misinterpret Tech Specs and incorrectly apply 3.0.3 since more than one system is inoperable; however, this would be incorrect since the individual LCOs apply.

B. **Incorrect**: The Unit is not in a shutdown statement.
Plausible: 24 hour Cold Shutdown statements are a part of the FRVS and EDG LCO action statements.

C. **Correct:** The "A" EDG is the limiting LCO. Continued operation is permitted for the next 72 hours. The mode switch is already in run, so the startup may continue since raising power is not an OPCON change as described in 3.0.4.

D. **Incorrect**: Tech Specs do not restrict power ascension in this instance.
Plausible: LCO 3.0.4 restricts OPCON changes when equipment is required to be operable in the next OPCON and there is a shutdown action associated with the LCO. In this case, the mode switch is already in RUN and raising power alone is NOT considered to be an OPCON change.

Level: SRO Exam
CFR 55.43(b)(2)

Lesson Plan Objective: TECSPCE010
Given specific plant operating conditions and a copy of the Hope Creek Generating Station Technical Specifications, evaluate plant/system operability and determine required actions (if any) to be taken. (**SRO/STA Only**)

Source: INPO Bank (HC Question 08/10/1998)

- Replaced distracter (D) with a new distracter.
- Modified distracter (A) to change time from 24 to 12 hours to Hot Shutdown since this would make LCO 3.0.3 plausible.

Level of knowledge: Analysis

Reference(s):
Tech Specs (3.03, 3.0.4, EDG, and FRVS)

KA: 264000.G2.2.22
2.2.22 Knowledge of Limiting Conditions for Operations and safety limits. (CFR: 43.2 / 45.2) 4.1

Comments / Change Record:

- None

Question: 94 Answer: B

1 Pt(s) Given the following:

- The Unit was shutdown on the previous shift to begin a refueling outage.
- Reactor temperature is currently 185°F with the reactor head installed and fully tensioned.
- The "A" EDG is out of service for modification work.
- The MODE switch has been placed in STARTUP/HOT STANDBY for RPS testing.

What of the following describes the current Operational Condition and related requirements?

A. Startup/Hot Standby and operation in this condition is allowed provided control rods are verified to be fully inserted.

B. Cold Shutdown and operation in this condition is allowed provided control rods are verified to be fully inserted.

C. Cold Shutdown and all requirements for being in Operational Condition 2 must be met.

D. Startup/Hot Standby and all requirements for being in Operational Condition 2 must be met.

Distracter Analysis:

A. **Incorrect**: Startup/Hot Standby is a mode switch position and not an Operational Condition. OPCON 2 is the "Startup" OPCON and transitioning from OPCON 4 to OPCON 2 is NOT permitted with an EDG out of service.
Plausible: The mode switch is permitted to be placed in Startup/Hot Standby while in OPCON 4 (Cold Shutdown) if all rods are verified to be fully inserted.

B. **Correct:** The Unit is in OPCON 4 (Cold Shutdown). The mode switch may be placed in Startup/Hot Standby for testing in this OPCON provided that the control rods are verified to remain fully inserted.

C. **Incorrect**: The requirements of OPCON 2 are NOT required to be met before placing the mode switch in Startup/Hot Standby to support RPS testing.

Plausible: The OPCON is correct (OPCON 4 Cold Shutdown).

D. **Incorrect**: Startup/Hot Standby is a mode switch position and not an Operational Condition. The requirements of OPCON 2 are NOT required to be met before placing the mode switch in Startup/Hot Standby to support RPS testing.
Plausible: Startup/Hot Standby is the correct mode switch position for OPCON 2.

Level: SRO Exam

Lesson Plan Objective: TECSPCE005
Define or discuss the terms contained in Section 1.0 of Hope Creek Generating Station Technical Specifications.

Source: HC Bank (Q53260)
1997 NRC Exam

Level of knowledge: Memory

Reference(s):
Tech Specs

KA: G2.1.22
2.1.22 Ability to determine Mode of Operation. (CFR: 43.5 / 45.13)
IMPORTANCE RO 2.8 SRO 3.3

Comments / Change Record:

- None

Question: 95 Answer: D

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1 Pt(s) The reactor is operating at 100% power. The licensed operator "at the controls" is permitted to go to:

A. the shift clerk area to make a copy of the FRVS Monthly Surveillance Test.

B. the Neutron Monitoring System back panels (10C608) to second verify red blocking tags.

C. the SM office to brief the SM on plant conditions during a plant fire.

D. the RM-11 to respond to a radiation monitor alarm.

Distracter Analysis:

A. Incorrect: See D.
B. Incorrect: See D.
C. Incorrect: See D.
D. Correct: IAW NC.NA-AP.ZZ-0005 Att #1 the RM-11 is the only area considered "At the Controls". Also reference HC.OP-AP.ZZ-0005.

Level: SRO Exam
CFR 55.43(b)(2)

Lesson Plan Objective: ADMPRO5CE004
Given a copy of the control room layout Identify the area denoted "at the controls," IAW NC.NA-AP.ZZ-0005

Source: HC Bank (Question 56186)

Level of knowledge: Memory

Reference(s):
NC.NA-AP.ZZ-0005 Att #1
HC.OP-AP.ZZ-0005

KA: G2.1.4
2.1.4 Knowledge of shift staffing requirements.
(CFR: 41.10 / 43.2) IMPORTANCE RO 2.3 SRO 3.4

Comment / Change Record:

- None

Question: 96 Answer: C

1 Pt(s) In accordance with site administrative processes, a 10CFR 50.59 Evaluation determines if a proposed change, test or experiment requires:

A. NRC notification prior to performance.

B. Inspection by the NRC during the activity.

C. Prior NRC approval via a license amendment.

D. Evaluation for compliance with NRC Reg Guides.

Distracter Analysis:

A. Incorrect: 50.59 evaluates for license amendment requirements not notification requirements.
B. Incorrect: NRC determines inspection requirements.
C. Correct: The evaluation determines if a license amendment is required.
D. Incorrect: Does not evaluate compliance with NRC Reg Guides.

Level: SRO Exam
CFR 55.43(b)(3)

Lesson Plan Objective: ADMPROE078
From Memory State the purpose of 10CFR50.59 Reviews and Safety Evaluations. IAW NC.NA-AP.ZZ-0059.

Source: HC Bank (Q77724)

Level of knowledge: Memory

Reference(s):
LS-AA-104

KA: G2.2.5
2.2.5 Knowledge of the process for making changes in the facility as described in the safety analysis report. (CFR: 43.3 / 45.13)
IMPORTANCE RO 1.6 SRO 2.7

Comments / Change Record:

- Replaced the words "Regulations" with "Reg Guides" to make the distracter even MORE incorrect. (licensee comment 09/11)

Question: 97 Answer: A

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1 Pt(s) Given the following:

- An I&C Technician is in the middle of SRM "A" Channel Functional Test
- The next section of his procedure contains several discrepancies

Which one of the following changes is PROHIBITED as an "On The Spot Change" to the procedure?

A. Raising the trip setpoint tolerance to reduce nuisance alarms.

B. Adding clarifying remarks to a procedural step.

C. Changing a step which corrects a switch nomenclature error.

D. Adding a supervisory review signoff.

Distracter Analysis:

A. **Correct:** Raising the tolerance of the trip setpoint is a change of intent because it is not being performed to align with Technical Specifications.
B. **Incorrect**: Clarifying a step is permitted under Attachment 1.
C. **Incorrect**: Correcting typographical errors is permitted under Attachment 1.
D. **Incorrect**: Changing the level of oversight is permitted IF the change results in increased oversight under Attachment 1.

Level: SRO Exam
CFR 55.43(b)(3)

Lesson Plan Objective: ADMPROE002
From Memory Describe what requirements must be satisfied to make an On-the-Spot change, and the required approval signatures.
IAW NC.NA-AP.ZZ-0001and NC.DM-AP.ZZ-0002.

Source: HC Bank (Q68868)
NRC Exam 2002

Level of knowledge: Memory

Reference(s):
NC.NA-AP.ZZ-0001and NC.DM-AP.ZZ-0002

KA: G2.2.11
Knowledge of the procedure for controlling temporary changes.
(CFR 41.10 / 43.3 / 45.13) IMPORTANCE RO 2.5 SRO 3.4

Comments / Change Record:

- None

Question: 98 Answer: B

1 Pt(s) Radiation Protection technicians have surveyed the Refuel Floor Reactor Head Laydown Area during an outage and obtained the following results:

- Area Dose Rates one foot from the source: 72 mr/hr
- Airborne Concentration: 0.15 DAC
- Smear Results: 750 dpm/100 cm2 gamma

Based on these results the area should be posted as a:

I. Radiation Area
II. High Radiation Area
III. Contaminated Area
IV. Airborne Radioactivity Area

A. I, III, and IV only

B. I and IV only

C. I and III only

D. II, III, and IV only

Distracter Analysis:

A. Incorrect: Not a Contamination Area since a Contamination Area= >1000 dpm/100cm2.
B. Correct: Radiation area = >5 mRem/hr to 100 mRem/hr. Airborne Radioactivity area = >10% or 10 DAC.
C. Incorrect: Not a Contamination Area since Contamination Area= >1000 dpm/100cm2.
D. Incorrect: Not a High Radiation Area since < 100 mRem/hr. Not a Contamination Area since Contamination Area= >1000 dpm/100cm2.

Level: SRO Exam
CFR 55.43(b)(4)

Lesson Plan Objective: NOH04ADM024C-02
From Memory State the definition of the following terms:

a. Contaminated Area
b. High Radiation Area
c. Locked High Radiation Area
d. Radiation Area
e. Restricted Area
f. Very High Radiation Area
g. Airborne Radioactivity Area
h. Declared Pregnant Woman (DPW)
i. Total Effective Dose Equivalent (TEDE) IAW NC.NA-AP.ZZ-0024

Source: Modified HC Bank (Q76884)
2002 LSRO Exam

Level of knowledge: Memory

Reference(s):
NC.NA-AP.ZZ-0024

KA: G2.3.1
Knowledge of 10 CFR: 20 and related facility radiation control requirements. (CFR: 41.12 / 43.4. 45.9 / 45.10)
IMPORTANCE RO 2.6 SRO 3.0

Comments / Change Record:

- Deleted "Very High Radiation Area" as a possible answer. Not needed and allowed a better psychometric balance and modified the question. (NRC reviewer 08/09)

Question: 99 Answer: A

1 Pt(s) Given the following:

- A Site Area Emergency was just declared 20 minutes ago due to a primary system line break that is discharging to the environment and it CANNOT be isolated from the control room.
- TSC & EOF personnel are beginning to arrive at their facilities; however, these facilities have NOT been fully activated at this time.
- Operators are developing a plan in the OSC to manually isolate the line in order terminate the release.
- An Emergency Dose Authorization will be needed to isolate the line.
- The Radiological Assessment Coordinator (RAC) has arrived in the TSC.
- The Emergency Duty Officer (EDO) has NOT arrived in the TSC, and the EDO CANNOT be reached by phone.

Who can authorize the Emergency Exposure in the absence of the EDO and what is the Planned Emergency Exposure Limit (PEEL)?

A. The Shift Manager is empowered to authorize up to 25 REM.

B. The Shift Manager is empowered to authorize up to 75 REM.

C. The RAC is empowered to authorize up to 25 REM.

D. The RAC is empowered to authorize up to 75 REM.

Distracter Analysis:

A. **Correct:** NC.EP-EP.ZZ-0304 states that the Shift Manager has the responsibility to authorize Emergency Exposures until the EDO assumes his or her responsibilities. The Planned Emergency Exposure Limit is 25 REM for accident mitigation and isolating the line is an accident mitigation action.

B. **Incorrect**: 75 REM is the PEEL limit for saving a life. Isolating the line is NOT a life saving action.

C. **Incorrect**: The RAC is NOT authorized to grant permission for an Emergency Exposure.

D. **Incorrect**: The RAC is NOT authorized to grant permission for an Emergency Exposure. 75 REM is the PEEL limit for saving a life. Isolating the line is NOT a life saving action.

Level: SRO Exam
CFR 55.43(b)(4) & (5)

Lesson Plan Objective: ????

Source: New

Level of knowledge: Memory

Reference(s):
NC.EP-EP.ZZ-0304, "Operational Support Center (OSC) Radiation Protection Response"

KA: G2.4.38
2.4.38 Ability to take actions called for in the facility emergency plan / including (if required) supporting or acting as emergency coordinator. (CFR: 43.5 / 45.11) IMPORTANCE RO 2.2 SRO 4.0

Comment / Change Record:

- None

Question: 100 Answer: D

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1 Pt(s) Given the following:

- You have received an authentic notification that a large commercial airliner departed from Philadelphia International Airport a few minutes ago and then diverted from its assigned flight plan.
- The airliner is NOT responding to Air Traffic Control.
- The airliner is flying at low altitude at about 500 feet above ground level and is heading towards the Salem/Hope Creek complex and it is closing at 250 mph.
- The airliner has been determined to be a credible threat to the station and it is less than 12 minutes away.

Which one of the following plant page scripts is correct for this situation?

Attention all personnel, Attention all personnel. A credible security threat exists at the station........

A. **Communicators report to the Control Room. All onshift personnel report to the OSC. All onsite ERO responders report to the NOSF immediately. All other personnel place your work area in a safe condition; exit the protected area immediately and proceed home.**

B. **Communicators report to the Control Room. All other personnel place your work area in a safe condition; take immediate cover and shelter in place until further notice.**

C. **All ERO members immediately report to your emergency response facility. All other personnel place your work area in a safe condition, exit the protected area immediately and proceed home.**

D. **Primary communicator report to the Control Room. All on shift personnel and onsite ERO responders report to the NOSF immediately. All other personnel place your work area in a safe condition; take immediate cover inside the nearest building and away from the containment building.**

Distracter Analysis:

A. **Incorrect**: *This is the script for an airborne threat* when there is greater than 15 minutes lead time.

B. **Incorrect**: This is the script for a NON airborne security threat with less than 15 minutes lead time.

C. **Incorrect**: This is the script for a NON airborne security threat with greater than 30 minutes lead time.

D. **Correct:** This is the correct script for an imminent airborne threat with less than 15 minutes lead time. See NC.EP-EP.ZZ-0102.

Level: SRO Exam
CFR 55.43(b)(5)

Lesson Plan Objective: ????

Source: New

Level of knowledge: Comprehension

Reference(s):
NC.EP-EP.ZZ-102

KA: G2.4.43
2.4.43 Knowledge of emergency communications systems and techniques. (CFR: 45.13) IMPORTANCE RO 2.8 SRO 3.5

Comments / Change Record:

- Moved lead in statement into the stem to reduce redundant words. (NRC reviewer)