Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 1 | Group # | 1 | |
| | K/A # | 007 Reactor Trip-Stabilization<br>EK1 Knowledge of the operational implications of the following concepts as they apply to the Reactor Trip:<br>EK1.06 Relationship of emergency feedwater flow to S/G and decay heat removal following reactor trip. | |
| | Importance Rating | 3.7 | |

Proposed Question:
Given the following conditions:

- The reactor tripped from 50% power.
- The crew has entered ES-0.1, 'Reactor Trip Response'.
- RCS temperature is 550°F and decreasing slowly.
- RCS pressure is stable at 2225 psig.

What is the cause of the RCS temperature decrease?

A. Safety Injection has actuated.

B. Steamline rupture in one of the steamlines.

C. High EFW flow coupled with low decay heat.

D. One of the turbine control valves has remained open.

Proposed Answer: C

A is incorrect but plausible. A safety injection could cause an RCS cooldown, however there if there were a safety injection the crew would not have transitioned to ES-0.1, 'Reactor Trip Response'.

B is incorrect but plausible. A steamline rupture would cause an RCS cooldown, however the cooldown would be more severe and there would also be an associated loss of RCS pressure.

C is correct. High EFW flow coupled with low decay heat will cause a slow RCS cooldown. Step 1 of ES-0.1, 'Reactor Trip Response' addresses RCS temperature control. The step includes the action to throttle EFW flow to maintain >500 gpm in the event that RCS temperature is less than 557°F and decreasing.

D is incorrect but plausible. A stuck open control valve could cause an RCS temperature decrease, however this would require a steam flowpath through the main steam isolation valves and turbine stop valves. E-0, 'Reactor Trip or Safety Injection' includes the immediate action to check the turbine tripped. The step specifically checks the stop valves closed, and if not, then the operator is directed to close the main steam isolation valves.

Technical Reference(s): ES-0.1, Reactor Trip Response

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 007 Reactor Trip-Stabilization |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.8/41.10/45.3 |
| Learning Objective: | L1225I13 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 1 | |
| Question 2 | Group # | 1 | |
| | K/A # | 009 Small Break LOCA<br>EK1 Knowledge of the operational implications of the following concepts as the apply to the small break LOCA:<br>EK1.02 Use of steam tables. | |
| | Importance Rating | 3.5 | |

Proposed Question:
Given the following conditions:

- A small break LOCA has occurred.
- RCS Wide Range pressure has stabilized at 1535 psig.
- Maximum Average Quadrant CETC temperature is 550°F.
- Highest Wide Range $T_{hot}$ is 530°F.

What is the expected Subcooling Monitor reading and the condition of the RCS?

A. +50°F, subcooled

B. -50°F, superheated

C. +70°F, subcooled

D. -70°F, superheated

Proposed Answer: A

A is correct. The subcooling monitor calculates subcooling based on RCS wide range pressure and average quadrant core exit thermocouple temperature. At 1535 psig there is approximately 50°F subcooling.

B is incorrect but plausible. The associated subcooling value derived from the steam tables is 50°F, however it is a subcooled condition.

C is incorrect but plausible. This would be the correct numerical value if the subcooling monitor utilized wide range loop temperature.

D is incorrect but plausible. This would be the correct numerical value if the subcooling monitor utilized wide range loop temperature.

Technical Reference(s): Steam tables

Proposed references to be provided to applicants during examination: Steam tables

| | |
|---|---|
| K/A Topic: | 009 Small Break LOCA |
| Question Source: | Modified from Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.8/41.10/45.3 |
| Learning Objective: | L8058I13 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 1 | |
| Question 3 | Group # | 1 | |
| | K/A # | 011 Large Break LOCA EK3 Knowledge of the reasons for the following responses as they apply to the Large Break LOCA: EK3.12 Actions contained in EOP for emergency LOCA(Large Break) | |
| | Importance Rating | 4.4 | |

Proposed Question:

The following plant conditions exist:

- A large LOCA has occurred.
- The crew has transitioned from E-0, 'Reactor Trip or Safety Injection' to E-1, 'Loss of Reactor or Secondary Coolant'.
- There is a RED PATH on the Core Cooling CSF.
- The following alarms have just actuated:
  - D4931 ECCS & CBS RECIRC INITIATED
  - D7193 RWST LEVEL LO-LO

What actions should the crew perform next?

A. Perform ES-1.3, 'Transfer to Cold Leg Recirculation' to completion and then transition to procedure and step in effect.

B. Immediately commence filling the RWST from any source while performing FR-C.1, 'Response to Inadequate Core Cooling'.

C. Perform FR-C.1, 'Response to Inadequate Core Cooling' steps until directed to return to procedure and step in effect, then perform ES-1.3, 'Transfer to Cold Leg Recirculation'.

D. Monitor CBS and RHR pump performance for signs of cavitation while performing FR-C.1, 'Response to Inadequate Core Cooling', and transfer to ES-1.3, 'Transfer to Cold Leg Recirculation', when the RWST EMPTY alarm annunciates.

Proposed Answer: A

A is correct. The Operator Action Summary Page of E-1 states "Go to ES-1.3, 'Transfer to Cold Leg Recirculation', step 1 if ECCS automatic switchover to containment sumps is actuated OR RWST level decreases to less than 115,000 gallons.

B is incorrect but plausible. Procedural rules of usage generically dictate transitioning to the appropriate functional restoration procedure in the event of a valid red path, however the procedural direction to proceed to ES-1.3 overrides this rule. Additionally, commencing an RWST makeup is a plausible solution, but is an applicable procedural direction if Emergency Coolant Recirculation capability was completely lost, and the crew were applying procedure ECA-1.1, Loss of Emergency Coolant Recirculation.

C is incorrect but plausible. Procedural rules of usage generically dictate transitioning to the appropriate functional restoration procedure in the event of a valid red path, however the procedural direction to proceed to ES-1.3 overrides this rule. ES-1.3 should be executed prior to implementing FR-C.1.

D is incorrect but plausible. It is prudent to monitor for signs of pump cavitation while implementing ES-1.3. The RWST EMPTY alarm is not an entry condition for ES-1.3, but is rather an OAS item that directs securing pumps taking suction from the RWST upon receiving the RWST EMPTY alarm. There is no procedural direction to transition from FR-C.1 to ES-1.3 based on this alarm.

| | |
|---|---|
| Technical Reference(s): | E-1, Loss of Reactor or Secondary Coolant |
| | ES-1.3, Transfer to Cold Leg Recirculation |
| | FR-C.1, Response to Inadequate Core Cooling |
| | ECA-1.1, Loss of Emergency Coolant Recirculation |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 011 Large Break LOCA |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/41.10/45.6/45.13 |
| Learning Objective: | L1203I06 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 1 | |
| Question 4 | Group # | 1 | |
| | K/A # | 015 RCP Malfunction AA2 Ability to determine and interpret the following as they apply to the Reactor Coolant Malfunctions: AA2.01 Cause of RCP Failure | |
| | Importance Rating | 3.0 | |

Proposed Question:
Which of the following would cause Reactor Coolant Pump #1 seal leakoff flow to indicate LOW and #2 seal leakoff flow to indicate HIGH?

A. Failure of the #1 seal.

B. Failure of the #2 seal.

C. Failure of the #3 seal.

D. Failure of the #1 seal followed by failure of the #2 seal.

Proposed Answer: B

A is incorrect but plausible. If the #1 seal failed then #2 seal flow and associated leakoff flow could increase, however there would be more flow through the #1 seal, so it's leakoff flow would indicate high.

B is correct. If the #2 seal fails then #1 seal flow would be directed to the #2 seal. #1 seal leakoff flow would decrease and #2 seal leakoff flow would increase.

C is incorrect but plausible. The #3 seal will not affect operation of the #1 or #2 seal.

D is incorrect but plausible. #1 seal failure would result in high #1 seal leakoff.

Technical Reference(s): Westinghouse RCP Manual

Proposed references to be provided to applicants during examination: None

K/A Topic: 015 RCP Malfunction

Question Source: Bank **2007 NRC Exam**

Question Cognitive Level: Higher: Comprehension/Analysis

10 CFR Part 55 Content: 43.5/45.13

Learning Objective: L1181I02

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 5 | Group # | 1 | |
| | K/A # | 026 Loss of Component Cooling<br>2.4.6 Knowledge of EOP Mitigation Strategies | |
| | Importance Rating | 3.7 | |

Proposed Question:
Given the following conditions:

- A LOCA has occurred.
- The crew is implementing E-0, 'Reactor Trip or Safety Injection'.
- Containment pressure is 23 psig.
- Reactor Coolant System pressure is 1320 psig.
- The highest Critical Safety Function is an ORANGE path on Containment (Z)
- All systems and automatic actuations have occurred as expected.

What action should be taken?

A. The Reactor Coolant Pumps should be stopped because pump cooling is lost.

B. The Reactor Coolant Pumps should be stopped per Adverse Containment criteria.

C. The Reactor Coolant Pumps should **NOT** be stopped because Reactor Coolant System pressure is above the trip setpoint.

D. The Reactor Coolant Pumps should **NOT** be stopped because they are required for forced cooling for these conditions.

Proposed Answer: A

A is correct. Per E-0 RCP trip criteria, the pumps should be secured if cooling flow is lost. If containment is at 23 psig then a 'P' signal will have isolated PCCW flow to the RCP's.

B is incorrect but plausible. The RCP's are secured due to the 'P' signal, which is actuated based on elevated containment pressure, however there are no adverse containment criteria that directs securing reactor coolant pumps.

C is incorrect but plausible. There are RCP trip criteria based on inadequate subcooling, however the trip criteria is not based on RCS pressure alone.

D is incorrect but plausible. It would be desirable to have forced cooling for LOCA conditions, provided the pump trip criteria (subcooling and pump cooling) are satisfied. In this case pump cooling has been isolated and the pumps should be stopped.

| | |
|---|---|
| Technical Reference(s): | E-0, Reactor Trip or Safety Injection. |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 026 Loss of Component Cooling |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.10/45.13 |
| Learning Objective: | L1202I03 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 6 | Group # | 1 | |
| | K/A # | 027 Pressurizer Pressure Control System Malfunction<br>AK2.03 Knowledge of the interrelationships between the Pressurizer Pressure Control Malfunctions and the following:<br>AK2.03 Controllers and positioners. | |
| | Importance Rating | 2.6 | |

Proposed Question:
Given the following conditions:

- The plant is at 100% power
- Pressurizer pressure is 2235 psig.
- Reactor Coolant System Temperature is 589°F.
- Pressurizer pressure channel PT-455 is the controlling channel.
- Pressurizer pressure channel PT-455 fails LOW.

With no operator action how will the plant respond?

A. All Pressurizer heaters will energize. No PORV will open.

B. All Pressurizer heaters will energize. Only PORV PCV-456A will open.

C. All Pressurizer heaters will energize. Only PORV PCV-456B will open.

D. All Pressurizer heaters will energize. Both PORV-456A and PORV-456B will open.

Proposed Answer: C

A is incorrect but plausible. It is true that the 'A' PORV will not function as PT-455 is the selected input to that controller, however pressure will rise and the 'B' PORV will still receive a valid pressure input signal from the backup pressure control channel and a valid arming signal from PT-457.

B is incorrect but plausible. The pressurizer heaters will energize causing pressure to rise however the 'A' PORV will not open as it is receiving an errant signal from PT-455.

C is correct. The heaters will receive a demand signal to energize. Additionally, the 'B' PORV will still receive a valid pressure input signal from the backup pressure control channel and a valid arming signal from PT-457. With control and backup heaters energized from the instrument failure pressure will eventually rise to the PORV opening setpoint.

D is incorrect but plausible. The heaters will energize. The 'B' PORV will open, however the 'A' PORV will not function as PT-455 is the selected input to that controller.

Technical Reference(s): 1-NHY-509026, Pressurizer Pressure Control

Proposed references to be provided to applicants during examination: None

K/A Topic: 027 Pressurizer Pressure Control System Malfunction

Question Source: Bank

Question Cognitive Level: Higher: Comprehension/Analysis

10 CFR Part 55 Content: 41.7/45.7

Learning Objective: L8027I14

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 7 | Group # | 1 | |
| | K/A # | 029 ATWS<br>EA1 Ability to operate and monitor the following as they apply to ATWS:<br>EA1.14 Driving of control rods into the core. | |
| | Importance Rating | 4.2 | |

Proposed Question:
Given the following conditions:

- A valid trip signal actuated and the reactor did not trip.
- The reactor will not trip manually from the control board.
- The crew has entered FR-S.1, 'Response to Nuclear Power Generation/ATWS'.

Per FR-S.1, 'Response to Nuclear Power Generation/ATWS', which of the following actions should be performed FIRST?

A. Dispatch an operator to trip the reactor locally.

B. Verify the Emergency Feedwater Pumps Running.

C. Initiate an Emergency Boration of the Reactor Coolant System.

D. Verify control rods are being inserted in auto or manually insert control rods.

Proposed Answer: D

A is incorrect but plausible. Dispatching an NSO to locally trip the reactor is on the FR-S.1 Operator Action Summary Page but is not an immediate action. Dispatching the operators to locally trip the reactor is important but should be done once the immediate actions of FR-S.1 are complete.

B is incorrect but plausible. Verifying EFW flow is important in order to maintain a heat sink while the reactor is still producing power. Preserving the heat sink is an FR-S.1 basis priority. Verification of EFW flow is not done until after the immediate actions are completed.

C is incorrect but plausible. Emergency borating the RCS is vital if efforts to trip the reactor are unsuccessful. This step is not performed until after the immediate actions of FR-S.1 are compete.

D is correct. The immediate action Step 1 RNO directs verifying control rods inserting in auto or manually inserting control rods.

Technical Reference(s): FR-S.1, 'Response to Nuclear Power Generation/ATWS'

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |
| K/A Topic: | 029 ATWS |
| Question Source: | New |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.7/45.5/45.6 |
| Learning Objective: | L1200I01, L1200I02, L1200I03 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 8 | Group # | 1 | |
| | K/A # | 038 Steam Generator Tube Rupture<br>2.2.25 Knowledge of bases in Technical Specifications for limiting conditions for operation and safety limits. | |
| | Importance Rating | 3.2 | |

Proposed Question:
What is the basis for the Technical Specification limit of 150 gallons per day primary to secondary leakage through any one steam generator?

A. The limit allows for leakage that does not interfere with detection of unidentified leakage. The leak rate criterion is well within the capability of the RCS Makeup System.

B. The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The leak rate criterion contributes to minimizing the frequency of steam generator tube ruptures.

C. The limit is based on operating experience with minimum detection capability of the main steam line radiation monitoring equipment. The leak rate criterion contributes to allowing for early detection of a potential offsite dose pathway.

D. The limit allows for early detection of a possible LOCA condition that bypasses containment. This limitation ensures that in the event of a LOCA safety injection inventory within containment will not be less than assumed in the safety analysis for emergency coolant recirculation.

Proposed Answer: B

A is incorrect but plausible. It is plausible that the 150 gpd limit will eliminate interference with unidentified leakage detection however this answer describes the basis for identified leakage.

B is correct. The correct answer contains the Tech. Spec. basis wording for Primary to Secondary Leakage through any one SG.

C is incorrect but plausible. Detection of a steam generator tube leak/rupture includes utilizing main steam line radiation monitors. Additionally, offsite dose pathway is a particular concern with tube leaks/ruptures.

D is incorrect but plausible. It is plausible that a propagated steam generator tube leak would contribute to loss of emergency coolant inventory from within the containment boundary however this answer is not the basis for steam generator tube leakage.

Technical Reference(s): Technical Specification Bases,

Reactor Coolant System
Leakage, Primary to Secondary
Leakage Through Any One SG,
page B 3/4 4-12

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |
| K/A Topic: | 038 Steam Generator Tube Rupture |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.5/41.7/43.2 |
| Learning Objective: | L8021I17 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 9 | Group # | 1 | |
| | K/A # | 040 Steam Line Rupture-Excessive Heat Transfer AA1 Ability to operate and/or monitor the following as they apply to the Steam Line Rupture. AA1.09 Setpoints of main steam safety and PORV's | |
| | Importance Rating | 3.4 | |

Proposed Question:
Given the following conditions:

- The plant is initially at 100% power.
- A Main Steam system leak is detected inside the turbine building.
- The crew trips the reactor and actuates a Main Steam Isolation.
- All equipment operates as designed.
- No operator actions are taken.

At what temperature should the RCS Tavg stabilize at?

A. 550°F

B. 557°F

C. 561°F

D. 567°F

Proposed Answer: C

A is incorrect but plausible. 550°F is the setpoint for the Steam Dump P-12 interlock. If there were an excessive heat removal condition with steam dumps in service they would go closed at 550°F. With a Main Steam Isolation actuated the steam dumps are not in service.

B is incorrect but plausible. If the reactor were tripped and the steam dumps were still in service then temperature would stabilize at 557°F on the steam dump plant trip controller.

C is correct. With the Main Steam Isolation actuated the steam leak should be isolated and the steam dumps are isolated. In this case RCS temperature control would be provided by the Atmospheric Steam Dump Valves which are upstream of the Main Steam Isolation Valves. The ASDV setpoints are set at 1125 psig, which corresponds to a Tavg of approximately 561°F.

D is incorrect but plausible. 567°F is the temperature associated with the 1185# lifting setpoint of the first safety valve for each steam line.

Technical Reference(s): E-0, Reactor Trip or Safety Injection

Proposed references to be provided to applicants during examination: None

K/A Topic: 040 Steam Line Rupture-Excessive Heat Transfer

Question Source: New

Question Cognitive Level: Higher: Comprehension/Analysis

10 CFR Part 55 Content: 41.7/45.5/45.6

Learning Objective: L8041I03, L8041I04

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 10 | Group # | 1 | |
| | K/A # | 054 Loss of Main Feedwater<br>2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactive release control, etc. | |
| | Importance Rating | 4.0 | |

Proposed Question:
Given the following conditions:

- The reactor is tripped and SI is actuated.
- Steam Generator pressures:
  - SG 'A': 1100 psig
  - SG 'B': 1100 psig
  - SG 'C': 1150 psig
  - SG 'D': 1150 psig
- Steam Generator Narrow Range levels:
  - SG 'A': 30%
  - SG 'B': 40%
  - SG 'C': 40%
  - SG 'D': 35%
- EFW flow is throttled to 50 gpm per Steam Generator.
- Reactor Coolant System pressure is 1375 and slowly decreasing.
- Containment pressure is 5 psig and slowly increasing.

The crew is exiting E-0, 'Reactor Trip or Safety Injection' and is preparing to enter E-1, 'Loss of Reactor or Secondary Coolant'. The crew is evaluating Critical Safety Functions.

What action should the crew take next?

A. Enter E-1, 'Loss of Reactor or Secondary Coolant'.

B. Enter FR-H.5, 'Response to Steam Generator Low Level'.

C. Enter FR-H.1, 'Response to Loss of Secondary Heat Sink'.

D. Enter FR-H.2, 'Response to Steam Generator Overpressure'.

Proposed Answer: A

A is correct. There are adequate steam generator narrow range levels to satisfy the Heat Sink Status Tree and all other lower level Heat Sink Status Tree criteria are met. The crew should enter E-1, 'Loss of Reactor or Secondary Coolant'.

B is incorrect but plausible. Adequate Steam Generator level is typically a challenge post trip. Operators are used to observing narrow range levels off scale low and making decisions based on wide range levels. The H.5 entry condition are not met as the adverse containment entry conditions are all SG NR levels not greater than 15%.

C is incorrect but plausible. Feedwater flow is less than the 500 gpm, however it is due to operator action. There is still 500 gpm available. Additionally, FR-H.1 entry conditions are not met as there are adequate narrow range levels in the steam generators.

D is incorrect but plausible. Steam generator pressures are elevated above normal operating pressure, however the Critical Safety Function Status Tree criteria for entering FR-H.2 is any Steam Generator pressure greater than 1225 psig.

| | |
|---|---|
| Technical Reference(s): | Critical Safety Function Heat Sink Status Tree |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 054 Loss of Main Feedwater |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/43.5/45.12 |
| Learning Objective: | L1211I01, L1211I03, L1211I05, L1211I15 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 11 | Group # | 1 | |
| | K/A # | 056 Loss of Offsite Power AK3 Knowledge of the reasons for the following as they apply to the Loss of Offsite Power: AK3.02 Actions contained in the EOP for loss of offsite power. | |
| | Importance Rating | 4.4 | |

Proposed Question:
Given the following conditions:

- The reactor has tripped.
- A loss of offsite power has occurred.
- The crew is performing the immediate actions of E-0, 'Reactor Trip or Safety Injection'.
- Bus 5 is energized from the 'A' Emergency Diesel Generator.
- Bus 6 is de-energized.

Which of the following describes the required action and basis for that action?

A. Transition to ECA-0.0, 'Loss of All AC Power' because one of the emergency busses has lost all AC power.

B. Do not attempt to restore power to Bus 6 while performing E-0 because it will delay higher priority operator actions.

C. Attempt to restore power to Bus 6 while continuing with E-0 because it is desirable to have power to all AC emergency busses.

D. Attempt to restore offsite power to both emergency busses while continuing with E-0 because the procedure assumes that offsite power is available.

Proposed Answer: C

A is incorrect but plausible. The crew would transition to ECA-0.0 if both emergency busses were deenergized. E-0 assumes that at least one AC emergency bus is energized. In this case Bus 5 is energized, so staying in procedure E-0 is appropriate.

B is incorrect but plausible. The crew should proceed with E-0 and take further actions, however it is desirable to pursue re-energizing bus 6 as it is desirable to have both busses energized.

C is correct. Bus 5 is energized so there is no immediate need to re-energize Bus 6 prior to continuing on in the procedure. Per the procedure basis, it is desirable to have both emergency busses energized, so this action should be pursued while continuing with the procedure.

D is incorrect but plausible. There should be an attempt to restore power to the de-energized bus, however the energized bus meets the basis for having at least one emergency bus energized.

| | |
|---|---|
| Technical Reference(s): | Westinghouse background document for procedure E-0 |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 056 Loss of Offsite Power |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/41.10/45.6/45.13 |
| Learning Objective: | L1202I03 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 12 | Group # | 1 | |
| | K/A # | 057 Loss of Vital Instrument Bus<br>AA2 Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus:<br>AA2.06 AC instrument bus alarms from the inverter and alternate power source. | |
| | Importance Rating | 3.2 | |

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- All bistable lights for Protection Channel 1 illuminate.
- The following alarms are present:
  - D5800 Vital Instrument Panel 1A Power Lost
  - D6003 Vital UPS 1A AC Output Volts Low

Which of the following describes the expected plant response to these conditions?

A. CS-FK-121, Charging Flow Controller, fails to minimum output.

B. Reactor Trip actuation based on Steam Generator Low-Low Level.

C. Train 'A' PCCW temperature control valves fail to full cooling mode.

D. FW-PT-505 fails LOW. Control rods continuously insert if they are in AUTO.

Proposed Answer: D

A is incorrect but plausible. Loss of Vital Instrument Panel 1A does affect the charging. If the primary selected pressurizer level channel were LT-459 and PP-1A lost power then CS-FK-121 would receive a false low pressurizer level which would cause CS-FK-121 output to increase.

B is incorrect but plausible. A loss of Vital Instrument Panel 1A does affect Steam Generator level control, however the affect is that the level control valves selected for Channel 1 would slowly ramp open causing SG level to increase.

C is incorrect but plausible. A loss of 120 VAC would cause PCCW to fail to the full cooling mode however this would occur on a loss of Non-Vital PP-1E.

D is correct. FW-PT-505 fails low. Control Rods would continuously insert if they are in AUTO. Procedure OS1247.01, 'Loss of a 120 VAC Vital Instrument Panel (PP1A, 1B, 1C or 1D)', step 1 directs the operator to place control rods in manual to resolve this condition.

Technical Reference(s): OS1247.01, Loss of a 120 VAC Vital Instrument Panel (PP1A, 1B, 1C or 1D)

Proposed references to be provided to applicants during examination: None

K/A Topic: 057 Loss of Vital Instrument Bus

Question Source: Modified from bank

Question Cognitive Level: Higher: Comprehension/Analysis

10 CFR Part 55 Content: 43.5/45.13

Learning Objective: 1186I06, L1186I08

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 1 | |
| Question 13 | Group # | 1 | |
| | K/A # | 058 Loss of DC Power AK1 Knowledge of the operational implcations of the following concepts as they apply to Loss of DC Power: AK1.01 Battery charger equipment and instrumentation. | |
| | Importance Rating | 2.8 | |

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- An NSO is in the process of transferring Vital 125 VDC Bus 11A from its alternate battery supply to its normal battery supply.
- Due to operator error the NSO failed to note that battery charger 1-EDE-BC-1A was NOT connected to Vital 125 VDC Bus 11A.
- The NSO opened the Alternate Battery Supply Breaker to 125 VDC Bus 11A

Which of the following is a direct result of these conditions?

A. Loss of Feedwater Control

B. Loss of Main Turbine trip control.

C. Loss of Pressurizer PORV Block Valve control.

D. Loss of Main Feedwater Pump 'A' Emergency Oil Pump.

Proposed Answer: A

A is correct. Per OS1248.01, 'Loss of a Vital 125 VDC Bus' loss of DC Bus 11A or 11B will result in a loss of normal feedwater control. The conditions described in the stem result in loss of power to DC bus 11A.

B is incorrect but plausible. When the Main Turbine is off line the EHC system receives control power and tripping control power from the plants 125VDC system, however, when the Main Turbine is running the control power is supplied internally to the EHC system.

C is incorrect but plausible. PORV control power is lost, however block valve control is not affected.

D is incorrect but plausible. Loss of DC Bus 11A does interface with the feedwater system however it affects normal feedwater control. The feedwater pump emergency DC oil pumps are powered from non-vital 125V DC Bus 12A.

| | | |
|---|---|---|
| Technical Reference(s): | OS1248.01, Loss of a Vital 125 VDC Bus | ON1248.02, Loss of a Non-Vital 125 VDC Bus. |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 058 Loss of DC Power |
| Question Source: | Modified from Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.8/41.10/45.3 |
| Learning Objective: | L1189I02 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 14 | Group # | 1 | |
| | K/A # | 062 Loss of Nuclear Service Water<br>AK3 Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water:<br>AK3.03 Guidance actions contained in EOP for Loss of Nuclear Service Water. | |
| | Importance Rating | 4.0 | |

Proposed Question:
Procedure OS1216.01, 'Degraded Ultimate Heat Sink' has the operator shut down the affected Cooling Tower Pump if a boundary valve is open. What is the basis for taking this action?

A. The pump is stopped to prevent pump runout conditions.

B. The pump is stopped to prevent flooding of the turbine building.

C. The pump is stopped to minimize loss of tower inventory to the ocean.

D. The pump is stopped to prevent contamination of the tower with salt water.

Proposed Answer: C

A is incorrect but plausible. If a boundary valve is open then their would be an additional discharge path and resulting change in back pressure on the pump, however the procedure does not analyze this condition when checking boundary valve status.

B is incorrect but plausible. There are secondary side isolation valves (SW-V-4 and SW-V-5) that are checked closed, however the step is performed to preserve cooling tower inventory vice preventing turbine building flooding.

C is correct. Per OS1216.01, 'Degraded Ultimate Heat Sink' the Cooling Tower Pump is stopped if a boundary valve is open and tower level is decreasing. This is to prevent loss of tower inventory.

D is incorrect but plausible. If a boundary valve is open then there is an interface between the cooling tower loop and the ocean, , however the procedure does not analyze this condition when checking boundary valve status.

Technical Reference(s): OS1216.01, Degraded Ultimate Heat Sink

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |
| K/A Topic: | 062 Loss of Nuclear Service Water |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.4/41.8/45.7 |
| Learning Objective: | L1193I02 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 15 | Group # | 1 | |
| | K/A # | 065 Loss of Instrument Air AA1 Ability to operate and/or monitor the following as they apply to the Loss of Instrument Air: AA1.05 RPS | |
| | Importance Rating | 3.3 | |

Proposed Question:
The plant is at 100% power. A leak in the Instrument Air System occurs. The crew enters ON1242.01, 'Loss of Instrument Air'.

Given the following sequence of events:

1400 Instrument Air Pressure is decreasing at a moderate rate.

1412 Service Air pressure is 90 psig.

1434 The Feedwater Regulating Valves close.

1437 The Containment PCCW Isolation valves close

At what time is the crew required to trip the reactor?

A. 1412

B. 1434

C. 1437

D. 1447

Proposed Answer: B

A is incorrect but plausible. The procedure requires action to be taken when Service Air pressure drops below 90 psig, however the action is to isolate the service air header in an effort to stop the decrease in Instrument Air header pressure. With regard to reactor trip criteria, there is no defined air pressure value.

B is correct. Reactor Trip Criteria listed on the procedures OAS page includes "Loss of feedwater flow to SG(s). This is also called for at Step 5 of the procedure.

C is incorrect but plausible. The reactor is required to be tripped if PCCW cooling is isolated to containment, however the criteria sates "Within 10 minutes of losing PCCW flow to containment". Additionally, the need to trip the reactor based on loss of feedwater flow was reached first.

D is incorrect but plausible. The reactor is required to be tripped "Within 10 minutes of losing PCCW flow to containment" , however, the need to trip the reactor based on loss of feedwater flow was reached first.

| | |
|---|---|
| Technical Reference(s): | ON1242.01, Loss of Instrument Air |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 065 Loss of Instrument Air |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.5/45.6 |
| Learning Objective: | L1194I03 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 16 | Group # | 1 | |
| | K/A # | W/E04 LOCA Outside Containment<br>EK2 Knowledge of the interrelationships between LOCA Outside Containment and the following:<br>EK2.1 Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. | |
| | Importance Rating | 3.5 | |

Proposed Question:
Given the following conditions:

- Procedure ECA-1.2, 'LOCA Outside Containment' is in progress.
- RH-V-14, 'RHR Discharge to RCS' and RH-V-22, 'RHR Train 'A' Discharge Cross-Connect valves have been closed.
- RCS Wide Range Pressure instrumentation indicates that RCS pressure is continuing to decrease.

What can be determined with regard to the leak status?

A. The leak is NOT associated with the Train 'A' RHR discharge check valves.

B. The leak has been identified and isolated in the Train 'A' RHR system.

C. The leak has been identified as occurring in the Train 'B' RHR system.

D. The leak is non-isolable. The crew should transition to ECA-1.1, 'Loss of Emergency Coolant Recirculation'.

Proposed Answer: . A

A is correct. If RCS pressure is still decreasing then the piping failure is not associated with the Train 'A' RHR discharge piping.

B is incorrect but plausible. If the leak were in the Train 'A' RHR system then the actions taken per the question stem would have caused RCS pressure to stabilize.

C is incorrect but plausible. The actions taken in the question stem determine that the leak is not associated with Train 'A' RHR, however at this point the leak can not be determined to be specifically from Train 'B' RHR.

D is incorrect but plausible. The leak may still be isolable. Further actions may be successful in isolating the leak.

Technical Reference(s): ECA-1.2, LOCA Outside Containment

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | W/E04 LOCA Outside Containment |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.7 |
| Learning Objective: | L1209I06 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | RO | SRO |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 1 | |
| Question 17 | Group # | 1 | |
| | K/A # | W/E11 Loss of Emergency Coolant Recirculation EA1 Ability to operate or monitor the following as they apply to the Loss of Emergency Coolant Recirculation: EA1.2 Operating characteristics of the facility. | |
| | Importance Rating | 3.5 | |

Proposed Question:
Given the following conditions:

- ECA-1.1, 'Loss of Emergency Coolant Recirculation' is in progress.
- The crew is attempting to open CBS-V-8, 'Containment Sump Isolation Valve'.

What is preventing CBS-V-8, Train 'A' Containment Sump Isolation Valve from being opened?

A. RC-V-22 and RC-V-23, Reactor Coolant Suction Valves are OPEN.

B. CBS-V-2, Train 'A' RWST to RHR Suction Valve is CLOSED.

C. RH-V-35, Train 'A' RHR supply to SI/CCP Suction Valve is OPEN.

D. SI-V-89 and SI-V-90, SI Pump Recirc to RWST Isolation Valves are CLOSED.

Proposed Answer: A

A is Correct. RC-V-22 and 23 provide an interlock with CBS-V-8 such that they must be CLOSED in order to open CBS-V-8. Additionally, RC-V-22 and 23 would be out of configuration with regard to ECCS lineup for cold leg recirculation.

B is incorrect but plausible. CBS-V-2 is required to be in proper configuration for ECCS recirculation, and is checked at step 1 of ECA-1.1. CBS-V-2 does not have any interlock with CBS-V-8 opening circuitry.

C is incorrect but plausible. RH-V-35 should be open, as it would have been opened prior to this event when ECCS was swapped to Cold Leg Recirculation. RH-V-35 does not provide any interlock to the CBS-V-8 opening circuitry.

D is incorrect but plausible. SI-V-89 and 90 should be closed, as they would have been closed prior to this event when ECCS was swapped to Cold Leg Recirculation. SI-V-89 and 90 do not provide any interlock to the CBS-V-8 opening circuitry.

| | | |
|---|---|---|
| Technical Reference(s): | 1-NHY-503252 | ECA-1.1 Loss of Emergency Coolant Recirculation |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | W/E11 Loss of Emergency Coolant Recirc. |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.7/45.5/45.6 |
| Learning Objective: | L8035I03, L8033I03 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 18 | Group # | 1 | |
| | K/A # | W/E05 Inadequate Heat Transfer-Loss of Secondary Heat Sink EA2 Ability to determine and interpret the following as they apply to the Loss of Secondary Heat Sink: EA2.1 Facility conditions and the selection of appropriate procedures during abnormal and emergency conditions. | |
| | Importance Rating | 3.4 | |

Proposed Question:
Given the following conditions:

- The 'A' Emergency Diesel Generator is out of service.
- A Loss of Offsite Power occurs.
- The reactor is tripped.
- There is no EFW flow to the Steam Generators.
- The crew is implementing FR-H.1 'Response to Loss of Secondary Heat Sink'.
- Subsequently, the 'B' Emergency Diesel Generator trips.

What action should be taken?

A. Immediately transition to ECA-0.0, 'Loss of All AC Power'.

B. Remain in FR-H.1, 'Response to Loss of Secondary Heat Sink' until feed flow has been established, then transition to ECA-0.0, 'Loss of All AC Power'.

C. Remain in FR-H.1, 'Response to Loss of Secondary Heat Sink' until directed to return to procedure and step in effect, then transition to ECA-0.0, 'Loss of All AC Power'.

D. Remain in FR-H.1, 'Response to Loss of Secondary Heat Sink' AND perform Abnormal Operating Procedure OS1246.01, Loss of Offsite Power-Plant Shutdown in parallel.

Proposed Answer: A

A is correct. The Emergency Response Procedures and Functional Restoration Procedures assume at least one Emergency Bus is available. A transition should be made to ECA-0.0, 'Loss of All AC Power'.

B is incorrect but plausible. Establishing feed flow to the steam generators is a high priority item with regard to loss of heat sink conditions, however, there is no such procedural guidance provided, and the Emergency Response Procedures and Functional Restoration Procedures assume at least one Emergency Bus is available. A transition should be made to ECA-0.0, 'Loss of All AC Power'.

C is incorrect but plausible. FR-H.1 is a Red Path Functional Restoration procedure, which dictates remaining in the procedure until a transition step is reached, unless a higher priority Red Path Critical Safety Function is challenged. However, the Emergency Response Procedures and Functional Restoration Procedures assume at least one Emergency Bus is available. A transition should be made to ECA-0.0, 'Loss of All AC Power'.

D is incorrect but plausible. Abnormal procedures may be used in parallel with Emergency Operating Procedures as long as they do not hinder the process of the EOP. However, OS1246.01, 'Loss of Offsite Power-Plant Shutdown' is only applicable in Modes 5 and 6.

| | |
|---|---|
| Technical Reference(s): | ECA-0.0, 'Loss of All AC Power' |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | W/E05 Inadequate Heat Sink-Loss of Secondary Heat Sink |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 45.13 |
| Learning Objective: | L8067I02 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 19 | Group # | 2 | |
| | K/A # | 003 Dropped Control Rod AA1 Ability to operate and/or monitor the following as they apply to the Dropped Control Rod: AA1.02 Controls and components necessary to recover rod. | |
| | Importance Rating | 3.6 | |

Proposed Question:
During recovery of a dropped rod in Group 1 of Control Bank 'A' a Rod Control Urgent failure alarm is expected to actuate. What is the origin and cause of the alarm?

| | **ORIGIN** | **CAUSE** |
|---|---|---|
| A. | 1AC Power Cabinet | Multiplex Error |
| B. | 2AC Slave Cycler | Slave Cycler Failure |
| C. | 2AC Power Cabinet | Lift Current Regulation Failure |
| D. | 1AC Power Cabinet | Stationary Current Regulation Failure |

Proposed Answer: C

A is incorrect but plausible. The 1AC Power Cabinet is associated with Control Bank 'A', however the alarm is due to a Lift Current Regulation Failure due to Bank A Group 2 rods.

B is incorrect but plausible. The 2AC Power Cabinet is the cabinet that the alarm is associated with, however the alarm is not due to a Slave Cycler condition.

C is correct. A rod control urgent failure alarm will occur when a dropped rod is recovered in a two group bank. This alarm is caused by a lift regulation failure in the non-affected group.

D is incorrect but plausible. The 1AC Power cabinet is associated with Control Bank 'A', however the alarm is due to a Lift Current Regulation Failure in the non-affected group (group 2).

Technical Reference(s): OS1210.05, Dropped Rod

Proposed references to be provided to applicants during examination: None

K/A Topic: 003 Dropped Control Rod

Question Source: Bank

| | |
|---|---|
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.5/45.6 |
| Learning Objective: | L1185I05 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 20 | Group # | 2 | |
| | K/A # | 028 Pressurizer Level Malfunction<br>AK1 Knowledge of the operational implications of the following concepts as they apply to Pressurizer Level Control Malfunctions:<br>AK1.01 PZR reference leg abnormalities | |
| | Importance Rating | 2.8 | |

Proposed Question:
Given the following conditions:

- The plant is at 75% power.
- All controls are in automatic.
- Pressurizer level transmitter LT-459 is selected for control.
- LT-459 reference leg develops a slow leak.
- No operator actions are taken.

How do Pressurizer and Volume Control Tank level indications initially respond?

A. Actual pressurizer level decreases. Volume Control Tank Level increases.

B. Actual pressurizer level increases. Volume Control Tank Level increases.

C. Actual pressurizer level decreases. Volume Control Tank Level decreases.

D. Actual pressurizer level increases. Volume Control Tank Level decreases.

Proposed Answer: A

A is correct. If the reference leg has a leak then the variable leg will impose more pressure on the d/p device and cause the affected instrument to detect an increasing level. This condition will cause the Pressurizer Level Control system to respond by reducing charging flow. As charging flow decreases pressurizer level will decrease as inventory is still being removed via letdown. The decrease in charging flow combined with the unchanged letdown flow rate will cause Volume Control Tank level to increase.

B is incorrect but plausible. If the variable leg leaked then the affected instrument would detect a decreasing level. This condition would cause charging flow to increase, however, this would result in a decreasing Volume Control Tank level.

C is incorrect but plausible. If the reference leg has a leak then the variable leg will impose more pressure on the d/p device and cause the affected instrument to detect an increasing level. This condition will cause the Pressurizer Level Control system to respond by reducing charging flow. As charging flow decreases pressurizer level will decrease as inventory is still being removed via letdown. However, with reduced charging flow and unchanged letdown flow the Volume Control Tank level would be increasing.

D is incorrect but plausible. It is correct that Volume Control Tank level would decrease, however, a reference leg leak would cause a decreasing actual pressurizer level.

Technical Reference(s): 1-NHY-509027, Pressurizer Level Control Process Block Diagram

Proposed references to be provided to applicants during examination: None

K/A Topic: 028 Pressurizer Level Malfunction

Question Source: Modified from bank

Question Cognitive Level: Higher: Comprehension/Analysis

10 CFR Part 55 Content: 41.8/41.10/45.3

Learning Objective: L8027I01, L8027I05

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 21 | Group # | 2 | |
| | K/A # | 036 Fuel Handling Accident<br>AK2 Knowledge of the interrelationships between Fuel Handling Accidents and the following:<br>AK2.02 Radiation monitoring equipment (portable and installed) | |
| | Importance Rating | 3.4 | |

Proposed Question:
The following plant conditions exist:

- The plant is in MODE 5.
- The Containment Air Purge (CAP) system is operating in the Pre-Entry Purge Mode.
- RM-6535B goes in to High alarm and a Train 'B' Containment Ventilation Isolation (CVI) signal is actuated.

How does the CAP system respond?

A. Containment Pre-entry Purge supply fan (CAP-FN-9) will trip. Only Train 'B' isolation valves (CAP-V2 and V3) will close.

B. Containment Pre-entry Purge supply fan (CAP-FN-9) will <u>not</u> trip. Only Train 'B' isolation valves (CAP-V2 and V3) will close.

C. Containment Pre-entry Purge exhaust fan (CAP-FN-10) will <u>not</u> trip. All 4 isolation valves (CAP-V1, V2, V3 and V4) will close.

D. Containment Pre-entry Purge supply fan (FN-9) and Pre-Entry Purge exhaust fan (CAP-FN-10) will trip. Only Train 'B' isolation valves (CAP-V2 and V3) will close.

Proposed Answer: B

A is incorrect but plausible. CVI does trip fans and close valves however only "A" train trips fans.

B is correct. Only "A" train CVI trips fans. "B" train CVI only closes IRC valves, CAP-V2 and V3.

C is incorrect but plausible. CVI does close valves but “B” only closes IRC V2 and V3. The “A” train closes only ORC V1 and V4.

D is incorrect but plausible. CVI will trip fans and Close valves. Only “A” train trips fans.

| | | |
|---|---|---|
| Technical Reference(s): | 1-NHY-509048 | 1-NHY-503221 |
| 1-NHY-503222 | 1-NHY-503297 | |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | Fuel Handling Accident |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.7 |
| Learning Objective: | L8059I06 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 1 | |
| Question 22 | Group # | 2 | |
| | K/A # | 037 Steam Generator Tube Leak<br>AA1 Ability to operate and/or monitor the following as they apply to the Steam Generator Tube Leak<br>AA1.11 PZR level indication | |
| | Importance Rating | 3.4 | |

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- A Steam Generator tube leak has been identified on the 'A' Steam Generator.
- The crew has entered OS1227.02, 'Steam Generator Tube Leak'
- The Unit Supervisor directed the Reactor Operator to estimate a leak rate based on a CVCS flow rate balance.
- CVCS flow rates are:
  - ➢ Letdown flow: 80 gpm
  - ➢ Total Seal Return flow: 12 gpm
  - ➢ Charging flow: 120 gpm

Which of the following describes the control board action performed and the leak rate obtained?

A. The Reactor Operator adjusted charging and letdown flow as necessary to maintain Pressurizer level stable. The leak rate is 40 gpm.

B. The Reactor Operator adjusted charging and letdown flow as necessary to maintain Pressurizer level stable. The leak rate is 28 gpm.

C. The Reactor Operator adjusted charging and letdown flow as necessary to maintain Volume Control Tank level stable. The leak rate is 40 gpm.

D. The Reactor Operator adjusted charging and letdown flow as necessary to maintain Volume Control Tank level stable. The leak rate is 28 gpm.

Proposed Answer: B

A is incorrect but plausible. The correct method is to adjust charging and letdown flow as necessary to maintain Pressurizer level stable. With Pressurizer level stable the operator can then calculate the leak rate based on CVCS flow rate balance. The correct equation for calculating the leak rate is Charging Flow – (Letdown Flow + Total Seal Return Flow) = Leak Rate. The student would have choosen answer A if they did not account for seal return flow.

B is correct. The correct method is to adjust charging and letdown flow as necessary to maintain Pressurizer level stable. With Pressurizer level stable the operator can then calculate the leak rate based on CVCS flow rate balance. The correct equation for calculating the leak rate is Charging Flow – (Letdown Flow + Total Seal Return Flow) = Leak Rate.

C is incorrect but plausible. The correct method is to adjust charging and letdown flow as necessary to maintain Pressurizer level stable, not VCT level. The correct equation for calculating the leak rate is Charging Flow – (Letdown Flow + Total Seal Return Flow) = Leak Rate. The student would have choosen answer C if they did not account for seal return flow.

D is incorrect but plausible. The correct method is to adjust charging and letdown flow as necessary to maintain Pressurizer level stable, not VCT level.

Technical Reference(s): OS1227.02, Steam Generator Tube Leak

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 037 Steam Generator Tube Leak |
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.5/45.6 |
| Learning Objective: | L1180I05, L8024I09 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 23 | Group # | 2 | |
| | K/A # | 051 Loss of Condenser Vacuum<br>AK3 Knowledge of the reasons for the following as they apply to the Loss of Condenser Vacuum<br>AK3.01 Loss of steam dump capability upon loss of condenser vacuum. | |
| | Importance Rating | 2.8 | |

Proposed Question:
During a loss of condenser vacuum the following events are expected to occur (assuming no operator action):

1. Main Feedwater Pump(s) trip
2. Main Turbine trips
3. Standby Mechanical Vacuum Pump starts
4. Condenser Steam Dumps are blocked (C-9)

Which of the following correctly states the order in which these actions will occur on decreasing condenser vacuum?

A. 3,2,1,4

B. 4,3,2,1

C. 3,4,2,1

D. 4,3,1,2

Proposed Answer: C

A is incorrect but plausible. The first event listed is correct, as the design has the standby mechanical vacuum pump start first in order to mitigate the condition. The order of having the feed pumps trip after the turbine trips is correct. It is plausible to have the steam dump block occur last such that the dumps are available upon the turbine trip, however, the steam dumps are blocked below C-9 when condenser vacuum decreases to less than 25" in order to protect the main condenser.

B is incorrect but plausible. The order of having the feed pumps trip after the turbine trips is correct. The order of having the Steam Dump block as the highest priority is plausible as the C-9 setpoint is designed to protect the condenser on decreasing vacuum, however the mechanical vacuum pump auto start is the first action to take place.

C is correct. The actions occur as follows:

26" Standby Vacuum Pump starts

25" Loss of C-9, Steam Dumps blocked

22.4" Main Turbine trip

18.5" Main Feedpumps trip

D is incorrect. It is plausible for the steam dumps to block first followed by start of the standby air removal pump. This would pursue the priority of protecting the condenser and preserving vacuum.

Technical Reference(s): ON1233.01, Loss of Condenser Vacuum

Proposed references to be provided to applicants during examination: None

K/A Topic: 051 Loss of Condenser Vacuum

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.5/41.10/45.6/45.13

Learning Objective: L1188I06

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| | Tier # | 1 | |
| Question 24 | Group # | 2 | |
| | K/A # | W/E16 High Containment Radiation<br>EA1.1 Ability to operate and/or monitor the following as they apply to the High Containment Radiation:<br>EA1.1 Components and functions of control and safety systems including instrumentation, signals, interlocks, failure modes, and automatic and manual features. | |
| | Importance Rating | 3.1 | |

Proposed Question:
FR-Z.3, 'Response to High Containment Radiation Level', step 2, checks if the containment recirculation filter should be placed in service. Why is containment pressure verified less than 18 psig?

A. The 'P' signal will prevent the containment recirculation fans (FN-3A and FN-3B) from starting.

B. The 'P' signal will prevent the containment recirculation filter system realignment to the Filter Mode.

C. The radioactive release associated with a containment pressure of 18 psig exceeds the limits of the recirculation filter capability.

D. Containment pressure greater than 18 psig could damage the recirculation dampers when they are aligned to Filter Mode.

Proposed Answer: B

A is incorrect but plausible. The 'P' signal does provide an interlock with the system however it provides a start signal to the recirculation fans and prevents aligning to the Recirc Mode.

B is correct. The 'P' signal realigns dampers to the recirculation mode, bypassing the filter at 18 psig. This is to facilitate recirculation and mixing of the containment atmosphere.

C is incorrect but plausible. It is a common operator misconception that the 'P' signal prevents exceeding the capacity of the filters.

D is incorrect but plausible. It is plausible that pressure could damage the dampers however the system interlock/logic prevents this from happening. Furthermore it is not the basis for the interlock.

Technical Reference(s): FR-Z.3, Response to High Containment Radiation

Proposed references to be provided to applicants during examination: None

K/A Topic:

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.7/45.5/45.6

Learning Objective: L8038I04

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 1 | |
| Question 25 | Group # | 2 | |
| | K/A # | 067 Plant Fire On-Site AA2 Ability to determine and interpret the following as they apply to the Plant Fire. AA2.06 Need for pressurizing control room (recirculation mode). | |
| | Importance Rating | 3.3 | |

Proposed Question:
Given the following conditions:

- D7040, 'East Air Intake Smoke Concentration High' is in alarm.
- The Fire Brigade Leader reports that there is an oil fire directly adjacent to the Control Room Ventilation East Air Intake.
- The crew has entered OS1223.01, Loss of Control Room Ventilation or Air Conditioning.

Which of the following describes how the Control Room Ventilation system will be aligned to prevent smoke intrusion to the Control Room?

A. Control Room Ventilation will automatically align to FILTER RECIRC MODE. 1-CBA-V-9, East Air Intake Isolation Valve will automatically close.

B. Control Room Ventilation will automatically align to FILTER RECRIC MODE. 1-CBA-V-9, East Air Intake Isolation Valve must be manually closed.

C. Control Room Ventilation must be manually aligned to FILTER RECIRC MODE. 1-CBA-V-9, East Air Intake Isolation Valve will automatically close.

D. Control Room Ventilation must be manually aligned to FILTER RECIRC MODE. 1-CBA-V-9, East Air Intake Isolation Valve must be manually closed.

Proposed Answer: D

A is incorrect but plausible. Control Room Ventilation does receive an automatic FILTER RECIRCULATION signal on 2/2 detectors on any Radiation Monitor associated with the air intakes, however there is no automatic signal initiated by the air intake smoke detectors. Additionally, the East Air Intake Isolation Valve is required to be closed by procedure however the valve has no automatic features.

B is incorrect but plausible. Control Room Ventilation is required to be in FILTER RECIRCULATION mode, however this does not occur automatically. It is correct that the East Air Intake Isolation Valve must be manually closed.

C is incorrect but plausible. It is correct that Control Room Ventilation must be manually aligned to FILTER RECIRCULATION mode. Additionally, the East Air Intake Valve is required to be closed by procedure however the valve has no automatic features.

D is correct. Control Room Ventilation must be manually aligned to FILTER RECIRCULATION mode. Control Room Ventilation does receive an automatic FILTER RECIRCULATION signal on 2/2 detectors on any Radiation Monitor associated with the air intakes, however there is no automatic signal initiated by the air intake smoke detectors. The East Air Intake Valve is required to be closed by procedure however the valve has no automatic features.

| | |
|---|---|
| Technical Reference(s): | OS1223.01, Loss of Control Room Ventilation or Air Conditioning |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 067 Plant Fire On-Site |
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 43.5/45.13 |
| Learning Objective: | L8039I04, L8039I05, L1194I09 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| | Tier # | 1 | |
| Question 26 | Group # | 2 | |
| | K/A # | W/E15 Containment Flooding<br>EA2 Ability to determine and interpret the following as they apply to the Containment Flooding:<br>EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency conditions. | |
| | Importance Rating | 2.7 | |

Proposed Question:
Given the following conditions:

- There is a LOCA inside containment.
- The crew has entered E-1, 'Loss of Reactor or Secondary Coolant'.
- Critical Safety Function status is:
  - Subcriticality-Green
  - Core Cooling-Yellow
  - Heat Sink-Green
  - Integrity-Yellow
- Containment conditions are:
  - Containment pressure: 10 psig and slowly increasing
  - Containment building level: 6 feet and slowly increasing
  - Containment radiation: 7 R/hr and stable
  - All Containment Phase 'A' AND Phase 'B' penetrations are isolated.

What action should be taken?

A. Transition to FR-Z.1, 'Response to High Containment Pressure'.

B. Transition to FR-Z.2, 'Response to Containment Flooding'.

C. Transition to FR-Z.3, 'Response to Containment High Radiation'.

D. Remain in E-1, 'Loss of Reactor or Secondary Coolant'.

Proposed Answer: B

A is incorrect but plausible. Containment pressure is adverse and increasing, however, with the given plant conditions the criteria for entering FR-Z.1 would be Containment pressure >18 psig with an unisolated Phase 'A' or Phase 'B' penetration.

B is correct. The given plant conditions meet the FR-Z.2 criteria (Containment building level > 4.7 ft).

C is incorrect but plausible. The Critical Safety Function does include radiation level however the FR-Z.3 criteria is 10 R/hr or greater for a yellow path condition.

D is incorrect but plausible. FR-Z.2 is an orange path condition requiring transition from EOP network to address CSF challenge.

| | |
|---|---|
| Technical Reference(s): | FR-Z.2, Response to Containment Flooding |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | W/E15 Containment Flooding |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 45.13 |
| Learning Objective: | L1212I09 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 1 | |
| Question 27 | Group # | 2 | |
| | K/A # | W/E08 RCS Overcooling-PTS<br>2.4.18 Knowledge of specific bases for EOP's | |
| | Importance Rating | 3.3 | |

Which of the following describes how Reactor Coolant Pumps should be operated in FR-P.1, 'Response to Imminent Pressurized Thermal Shock' if SI cannot be terminated and why?

A. All RCP's should be secured to prevent small break LOCA concerns associated with an untimely RCP trip.

B. All but one RCP should be secured to reserve the pumps for operation later if forced core cooling is required.

C. An RCP should be started if possible to mix the incoming cold Safety Injection water with the warm Reactor Coolant.

D. All RCP's should be started to prevent stagnant loop cooldown resulting from a pressurized thermal shock event.

Proposed Answer: C

A is incorrect but plausible. This answer describes the basis for securing an RCP based on subcooling in the event of a small break LOCA. RCP's are not secured in FR-P.1 if they are running.

B is incorrect but plausible. This answer is plausible as it would leave a configuration of one RCP running, which is the goal of FR-P.1 however the procedure directs starting an RCP if they are not running. The procedure does not secure running RCP's.

C is correct per the Westinghouse basis document for FR-P.1.

D is incorrect but plausible. This answer is plausible since the desire is to mitigate the cooldown however the strategy is to start one RCP in order to mix the SI with the RCS.

Technical Reference(s): FR-P.1, Response to Pressurized Thermal Shock; Westinghouse Background Document, FR-P.1

Proposed references to be provided to applicants during examination: None

K/A Topic: W/E08 RCS Overcooling-PTS

Question Source: Bank

Question Cognitive: Memory or Fundamental Knowledge

| | |
|---|---|
| Level: | |
| 10 CFR Part 55 Content: | 41.10/43.1/45.13 |
| Learning Objective: | L1208I04 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 28 | Group # | 1 | |
| | K/A # | 003 Reactor Coolant Pump K2 Knowledge of bus power supplies to the following: K2.01 RCP's | |
| | Importance Rating | 3.1 | |

Proposed Question:
Which of the following correctly lists the electrical power supply and cooling water supply for Reactor Coolant Pump 1B?

A. 13.8kv Bus 1 and PCCW Train A

B. 13.8kv Bus 2 and PCCW Train A

C. 13.8kv Bus 1 and PCCW Train B

D. 13.8kv Bus 2 and PCCW Train B

Proposed Answer: C

A is incorrect but plausible. RCP 'B' is powered from 13.8kv Bus 1, however the cooling water is supplied from PCCW Train B. It is a common misconception that the 'A' and 'B' RCP's are cooled from Train B PCCW and the 'C' and 'D' RCP's are cooled from Train A PCCW.

B is incorrect but plausible. It is a common misconception that RCP 'B' is powered from 13.8kv Bus 2, as this logic is in alignment with the power supplies for safety related equipment, i.e., the 'A' SI pump is powered from Train 'A' (Bus 5). Additionally, it is a common misconception that the 'A' and 'B' RCP's are cooled from Train B PCCW and the 'C' and 'D' RCP's are cooled from Train A PCCW.

C is correct. 13.8kv Bus 1 supplies power to the 'A' and 'B' RCP's. Train B PCCW supplies cooling to the 'B' and 'C' RCP's.

D is incorrect but plausible. It is a common misconception that RCP 'B' is powered from 13.8kv Bus 2, as this logic is in alignment with the power supplies for safety related equipment, i.e., the 'A' SI pump is powered from Train 'A' (Bus 5).

Technical Reference(s): 1-CC-20210, PCCW Train B Overview; 1-NHY-310004, Station Main Electrical Bus One Line Diagram

Proposed references to be provided to applicants during examination: None

K/A Topic: 003 Reactor Coolant Pump

| | |
|---|---|
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.7 |
| Learning Objective: | L8021I33, L8036I04 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 29 | Group # | 1 | |
| | K/A # | 004 Chemical and Volume Control<br>K6 Knowledge of the effect of a loss or malfunction on the following CVCS components:<br>K6.07 Heat exchangers and condensers | |
| | Importance Rating | 2.7 | |

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- The Reactor Operator notices that Letdown flow is oscillating due to flashing downstream of the Letdown Flow Control Valves (CS-HCV-189 and 190).

Which of the following could have caused this condition?

A. The Regenerative Heat Exchanger has developed a tube leak.

B. CS-PCV-131, Letdown Pressure Control Valve, has failed closed.

C. PCCW flow to the Letdown Heat Exchanger has increased.

D. CS-FCV-121, Charging Flow Control Valve, has failed closed.

Proposed Answer: D

A is incorrect but plausible. If the Regenerative Heat Exchanger had a leak such that the letdown side depressurized there could be a flashing condition however the charging side of the heat exchanger is at a higher pressure. A tube leak would result in cooler charging flow leaking to the letdown side.

B is incorrect but plausible. CS-PCV-131 is a backpressure control valve. If the valve failed open then this could cause flashing.

C is incorrect but plausible. PCCW flow does have an affect on letdown temperature, however this heat exchanger is downstream of CS-PCV-131 and would have no affect on this condition.

D is correct. CS-FCV-121 regulates charging flow rate. Charging flow provides cooling in the Regenerative Heat Exchanger. If CS-FCV-121 failed close there would be a loss of cooling and flashing would occur.

Technical Reference(s): 1-NHY-506269, CS Regen HX 1-NHY-506279, CS Regen Return

1-NHY-506276, Charging Pump Control Loop

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 004 Chemical and Volume Control |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.7 |
| Learning Objective: | L1445I11 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 30 | Group # | 1 | |
| | K/A # | 004 Chemical and Volume Control<br>2.1.23 Ability to perform specific system and integrated plant procedures during all modes of operation. | |
| | Importance Rating | 3.9 | |

Proposed Question:
Given the following conditions:

- The Reactor Operator is placing CS-P-2A ('A' Charging Pump) in service per OS1002.02, 'Operation of Letdown, Charging, and Seal Injection'.
- The Reactor Operator has increased charging flow using CS-FCV-121, Charging Flow Control Valve.

What affect does this have on Seal Injection flow and what action is required to return Seal Injection flow to it's original value?

A. Seal Injection flow has DECREASED. CS-HCV-182, Seal Injection Flow Control Valve must be THROTTLED CLOSED.

B. Seal Injection flow has DECREASED. CS-HCV-182, Seal Injection Flow Control Valve must be THROTTLED OPEN.

C. Seal Injection flow has INCREASED. CS-HCV-182, Seal Injection Flow Control Valve must be THROTTLED CLOSED.

D. Seal Injection flow has INCREASED. CS-HCV-182, Seal Injection Flow Control Valve must be THROTTLED OPEN.

Proposed Answer: D

A is incorrect but plausible. It is a common misconception that raising charging flow will cuase seal injection flow to decrease. This would be the case if charging and seal injection flow paths were in parallel however CS-HCV-182 acts as a backpressure valve which directs a portion of charging flow to the seals. Raising charging flow will cause seal injection flow to increase.

B is incorrect but plausible. It is true that CS-HCV-182 should be THROTTLED OPEN, however, this is because charging flow would have INCREASED.

C is incorrect but plausible. It is true that seal injection flow would increase as charging flow is increased however CS-HCV-182 must be THROTTLED OPEN in order to reduce seal injection flow back to it's original value.

D is correct. Seal Injection flow will increase if charging flow is increased. CS-HCV-182 acts as a backpressure valve which forces a portion of charging flow to the seals. In order to reduce seal injection flow CS-HCV-182 should be THROTTLED OPEN.

Technical Reference(s): OS1002.02

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 004 Chemical and Volume Control |
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.10/43.5/45.2/45.6 |
| Learning Objective: | L8024I03 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 31 | Group # | 1 | |
| | K/A # | 005 Residual Heat Removal<br>K4 Knowledge of RHR design feature (s) and/or interlock (s) which provide the following:<br>K4.03 RHR heat exchanger bypass flow control. | |
| | Importance Rating | 2.9 | |

Proposed Question:
Given the following plant conditions:

- The Train 'A' Residual Heat Removal system is in service with flow set at 3500 gpm.
- Reactor Coolant System temperature is 300°F and decreasing.
- Reactor Coolant System pressure is 340 psig and stable.
- The current cooldown rate is 12°F/hr.
- The Unit Supervisor directs the Reactor Operator to increase the cooldown rate to 40°F/hr.

Which of the following describes how the operator will increase the cooldown rate?

A. THROTTLE CLOSED RHR-FCV-618, RHR Heat Exchanger Bypass Valve. Total system flow is automatically maintained by RHR-FCV-610, RHR Pump Mini Flow Valve.

B. THROTTLE CLOSED RHR-HCV-606, RHR Heat Exchanger Outlet Valve. Total system flow is automatically maintained by RHR-FCV-618, RHR Heat Exchanger Bypass Valve.

C. THROTTLE OPEN RHR-HCV-606, RHR Heat Exchanger Outlet Valve. Total system flow is automatically maintained by RHR-FCV-618, RHR Heat Exchanger Bypass Valve.

D. THROTTLE OPEN RHR-FCV-618, RHR Heat Exchanger Bypass Valve. Total system flow is automatically maintained by RHR-FCV-610, RHR Pump Miniflow Valve.

Proposed Answer: C

A is incorrect but plausible. It is conceivable that throttling the heat exchanger bypass would force more flow through the heat exchanger however total system flow is controlled by RHR-FCV-618 not RHR-FCV-610.

B is incorrect but plausible. Throttling the heat exchanger outlet valve close would allow the process fluid more time in the heat exchanger however the mass flow rate through the heat exchanger would decrease and cause a reduction in the cooldown rate.

C is correct. Throttling open RHR-CV-606 allows a higher mass flow rate through the heat exchanger. As flow rate through the heat exchanger is increase RHR-FCV-618 will decrease total system flowrate.

D is incorrect but plausible. It is conceivable that total system flow could be maintained by the pump mini-flow valve however system flow is maintained automatically by RHR-FCV-618.

Technical Reference(s): OS1013.03, Residual Heat Removal Train A Startup and Operation

Proposed references to be provided to applicants during examination: None

K/A Topic: 005 Residual Heat Removal

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.7

Learning Objective: L8033I06, L8033I07

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 32 | Group # | 1 | |
| | K/A # | 005 Residual Heat Removal 2.1.32 Ability to explain and apply system limits and precautions. | |
| | Importance Rating | 3.8 | |

Proposed Question:
Given the following plant conditions:

- The crew is performing OS1000.14, 'RCS Evacuation and Fill'.
- The operators are draining the Reactor Coolant System to minus 84.5 inches in preparation for evacuating the system.

What is the Residual Heat Removal cooling flow rate limit for these conditions and why?

A. Maximum flow rate of 1600 gpm to prevent Pressurizer surge line flooding.

B. Maximum flow rate of 1600 gpm to prevent air entrainment in the RHR suction line due to vortexing.

C. Maximum flow rate of 5000 gpm to allow for maximum cooling AND prevent Pressurizer surge line flooding.

D. Maximum flow rate of 5000 gpm to allow for maximum cooling AND prevent air entrainment in the RHR suction line due to vortexing.

Proposed Answer: B

A is incorrect but plausible. When the RCS system is evacuated there is less back pressure on the RHR pump. It is conceivable that this condition could cause RHR pump flow to increase, with fluid drawn into the pressurizer surge line. Minus 84.5 inches is the centerline of the surge line. This is not the basis for the flow limitation.

B is correct. OS1000.14, Section 6.4 lowers RCS level to minus 84.5 inches. The CAUTION statement associated with this procedure section states "The plant must be in MODE 5 and operating train RHR flow restricted to 1600 gpm before lowering water level." The concern is in regard to pump cavitation at reduced inventory. The 1600 gpm flow limitation is to protect against pump caviation in the event of a loss of instrument air.

C is incorrect but plausible. Core cooling is a priority. Also, minus 84.5 inches is the center of the surge line, which makes surge line flooding a plausible concern.

D is incorrect but plausible. The concern is air entrainment and pump vortexing however the flow rate restriction is 1600 gpm.

Technical Reference(s): OS1000.14, RCS Evacuation and Fill

Proposed references to be provided to applicants during examination: None

K/A Topic: 005 Residual Heat Removal

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.10/43.2/45.12

Learning Objective: L1166I04

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 33 | Group # | 1 | |
| | K/A # | 006 Emergency Core Cooling<br>A4 Ability to manually operate and/or monitor from the control room:<br>A4.11 Over pressure protection system. | |
| | Importance Rating | 4.2 | |

Proposed Question:
Given the following conditions:

- The plant is in MODE 4.
- Loss of pressure control results in a pressure excursion.
- F4388, 'RHR Suction Valve 23 Open & RCS Pressure High' is in alarm.
- Pressurizer Relief Tank level and pressure are both increasing.
- Primary Drain Tank pressure and level are stable.

Which of the following could cause these conditions to occur?

A. The alarm actuated when RCS pressure increased to greater than 362 psig. The Train 'A' RHR suction relief has lifted.

B. The alarm actuated when RCS pressure increased to greater than 600 psig. The Train 'A' RHR suction relief has lifted.

C. The alarm actuated when RCS pressure increased to greater than 362 psig. The Train 'A' RHR discharge relief has lifted.

D. The alarm actuated when RCS pressure increased to greater than 600 psig. The Train 'A' RHR discharge relief has lifted.

Proposed Answer: A

A is correct. The alarm setpoint is 362 psig. The suction relief valve lifts at 450 psig and is routed to the PRT.

B is incorrect but plausible. The suction relief is routed to the PRT however the alarm setpoint is 362 psig. 600 psig is plausible as this is the pressure associated with the RHR discharge piping relief valve. The discharge relief valve is routed to the PDT.

C is incorrect but plausible. The setpoint for the alarm is 362 psig however the RHR discharge piping relief valve is routed to the Primary Drain Tank.

D is incorrect but plausible. The 600 psig pressure is associated with the suction relief valve lifting setpoint. Additionally, the discharge relief valve is routed to the Primary Drain Tank

Technical Reference(s): VAS Procedure, F4388, RHR SUCTION Valve 23 Open and RCS Pressure Hi

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 006 Emergency Core Cooling |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.5 to 45.8 |
| Learning Objective: | L8033I02 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 34 | Group # | 1 | |
| | K/A # | 007 Pressurizer Relief/Quench Tank K3 Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: K3.01 Containment | |
| | Importance Rating | | |

Proposed Question:
Given the following plant conditions:

- The plant is at 100% power.
- One of the Pressurizer Power Operated Relief Valves (PORV's) is leaking past it's seat.
- The Pressurizer Relief Tank (PRT) cooling system does NOT function as designed.

Which of the following describes the expected result of these conditions if no operator action is taken?

A. PRT pressure rises causing the rupture disks to relieve pressure to the containment atmosphere.

B. PRT pressure rises causing the PRT relief valves to relieve pressure to the containment atmosphere.

C. PRT level rises causing water to be diverted through overflow lines to the containment building sump.

D. PRT level rises causing water to be automatically pumped to the Reactor Coolant Drain Tank.

Proposed Answer: A

A is correct. Without the normal PRT cooling system available, the PRT level would rise and pressure in the tank would continue to increase until rupture disks relieve tank pressure to containment atmosphere.

B is incorrect but plausible. The PRT pressure would relieve to containment atmosphere however the PRT does not have relief valves.

C is incorrect but plausible. PRT level would increase due to the PORV leak however there is no level divert flowpath to the containment building sump.

D is incorrect but plausible. PRT level would increase due to the PORV leak however a PRT pumpdown to the RCDT is an operator initiated action to reduce PRT level. Also, if

PRT water temperature is greater than 120°F a control interlock prevents operation of RC-V-309 to transfer water out of the PRT.

Technical Reference(s): PID-1-RC-B20846, Reactor Coolant System Pressurizer

Proposed references to be provided to applicants during examination: None

K/A Topic: 007 Pressurizer Relief/Quench Tank

Question Source: Bank

Question Cognitive Level: Higher: Comprehension/Analysis

10 CFR Part 55 Content: 41.7/45.6

Learning Objective: L8022I11

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 35 | Group # | 1 | |
| | K/A # | 008 Component Cooling Water<br>K1 Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems:<br>K1.02 Loads cooled by CCWS | |
| | Importance Rating | 3.3 | |

Proposed Question:
Given the following plant conditions:

- The plant was initially at 100% power.
- A LOCA has occurred.
- Safety Injection has actuated.
- Containment pressure peaked at 15 psig and is slowly decreasing.

Which of the following describes the status of Primary Component Cooling Water system components for these conditions?

A. PCCW Radiation Monitor Isolation Valves are CLOSED.

B. Excess Letdown Heat Exchanger PCCW Isolation Valve is OPEN.

C. Residual Heat Removal Heat Exchanger PCCW Isolation Valves are OPEN.

D. Containment Building Spray Heat Exchanger PCCW Isolation Valves are OPEN.

Proposed Answer: C

A is incorrect but plausible. The PCCW radiation monitor isolation valves do have an automatic isolation signal associated with them however the signal is generated from PCCW level.

B is incorrect but plausible. A 'T' signal would be generated for the given conditions. The 'T' signal will cause a normal Letdown isolation. It is plausible that the excess letdown heat exchanger isolation valve would open to support excess letdown operations in order to support further EOP actions, however, further EOP actions would be associated with returning normal letdown to service if conditions permited.

C is correct. The RHR heat exchanger PCCW isolation valves will automatically open on a 'T' signal.

D is incorrect but plausible. The CBS heat exchanger isolation valves do have an automatic open signal however this is generated by a 'P' signal which occurs when containment pressure reaches 18 psig. The conditions listed in the question stem state that containment pressure did not reach 18 psig.

| | | |
|---|---|---|
| Technical Reference(s): | 1-NHY-503271, CBS & RHR Outlet Valves Logic Diagram | 1- NHY-503275, Excess Letdown Heat Exchanger Outlet Valve 34 Logic Diagram |
| | 1-NHY-503281, PCCW Liquid Radiation Monitor Sample Flow Control Logic Diagram | |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 008 Component Cooling Water |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.2 to 41.9/45.7 to 45.9 |
| Learning Objective: | L8036I12 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 2 | |
| Question 36 | Group # | 1 | |
| | K/A # | 008 Component Cooling Water<br>A2 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including:<br>A2.01 Loss of CCW pump | |
| | Importance Rating | 3.3 | |

Proposed Question:
Given the following plant conditions:

- The plant is at 100% power.
- The 'A' Primary Component Cooling Water pump tripped on high header temperature.

Assuming no operator actions are taken and the high header temperature alarm does not clear, what is the status of Train 'A' PCCW 5 minutes after the 'A' pump tripped?

A. The 'C' PCCW pump is running.

B. The 'A' PCCW pump is running.

C. No Train 'A' PCCW pumps are running.

D. The 'A' and 'C' PCCW pumps are running.

Proposed Answer: C

A is incorrect but plausible. There is an auto start feature for the standby pump which has a time delay associated with it's circuit, however this is in the event that there were a loss of power condition. If there is a high temperature condition for greater than 60 seconds then both pumps are locked out from starting until the condition clears.

B is incorrect but plausible. The high temperature pump trip circuit has a 60 second time delay associated with it. It is plausible to associate this 60 seconds with a lockout such that the pump would restart after a 60 second delay.

C is correct. If there is a high temperature condition for greater than 60 seconds then both pumps are locked out from starting until the condition clears. The basis for this feature is that the PCCW piping seismic supports are not designed to withstand a safe shutdown earthquake at high temperature.

D is incorrect but plausible. It is plausible that the circuitry would automatically start both pumps after a time delay in order to increase system flow to combat the over temperature condition.

| | |
|---|---|
| Technical Reference(s): | 1-NHY-503270, PCCW Pumps Logic Diagram |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 008 Component Cooling Water |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/43.5/45.3/45.1<br>3 |
| Learning Objective: | L8036I07 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 37 | Group # | 1 | |
| | K/A # | 010 Pressurizer Pressure Control<br>K4 Knowledge of PZR PCS design features and/or interlocks which provide for the following:<br>K4.03 Over pressure control | |
| | Importance Rating | 3.8 | |

Proposed Question:
Given the following plant conditions:

- The plant is at 100% power.
- The Pressurizer Pressure Control System is in AUTOMATIC
- Pressurizer pressure channels PT-455/456 are selected for controlling and backup channels.

Which of the following correctly describes the function of Pressurizer pressure channel PT-458 with this channel alignment?

A. Opens RC-PCV-456A (PORV A) when pressure increases to 2385 psig.

B. Opens RC-PCV-456B (PORV B) when pressure increases to 2385 psig.

C. Opens RC-V-122, Block Valve for RC-PCV-456A (PORV A) when pressure increases to 2350 psig.

D. Opens RC-V-124, Block Valve for RC-PCV-456B (PORV B) when pressure increases to 2350 psig.

Proposed Answer: C

A is incorrect but plausible. PT-458 does provide a signal to the Train 'A' PORV circuitry however it arms the 'A' PORV and opens the 'A' PORV block valve versus providing an opening signal to the 'A' PORV.

B is incorrect but plausible. This would be correct if PT-458 were selected as the backup channel for pressure control. The selected backup channel will open the B PORV. For the conditions listed in the question this would be from PT-456.

C is correct. PT-458 provides an opening signal to RC-V-122, the A PORV Block Valve.

D is incorrect but plausible. PT 458 does provide a block valve opening signal, however it is associated with RC-V-122, the A PORV Block Valve.

Technical Reference(s): 1-NHY-509026, Pressurizer

Pressure Control Process Control
Block Diagram

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 010 Pressurizer Pressure Control |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7 |
| Learning Objective: | L8027I06 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 38 | Group # | 1 | |
| | K/A # | 012 Reactor Protection K6 Knowledge of the effect of loss or malfunction that the following will have on the RPS: K6.10 Permissive circuits | |
| | Importance Rating | 3.3 | |

Proposed Question:

The following plant conditions exist:

- The plant was initially at 52% power.
- A LOCA occurred resulting in dual train Safety Injection and Reactor Trip signals.
- Reactor trip breaker "B" (RTB) does not open, and cannot be opened by any means.
- All other systems and components function as designed.

What is the effect of reactor trip breaker 'B' failing to open?

A. Train "B" ECCS pumps will not automatically start.

B. Train "B" automatic Safety Injection signals can not be blocked.

C. Train "A" and Train "B" automatic Safety Injection signals can not be blocked.

D. Train "B" ECCS pumps will automatically start, then automatically stop 60 seconds later.

Proposed Answer: B

A is incorrect but plausible. The P-4 input feeds into the SI reset and block circuit which could plausibly block the SI signal from starting ECCS pumps.

B is correct. With "B" reactor trip breaker closed the blocking function of the reset and block switch actuation does not seal in. The existing SI signal will be reset but if another SI signal exists or actuates then the SI actuation will re-occur. Only for the "B" train.

C is incorrect but plausible. The P-4 input into the SI circuit is shown on a common "A" and "B' train drawing (1-NHY-509048). Notes in the drawing refer to two redundant trains. The actual drawing only shows one input and this could be interpreted as affecting both trains.

D is incorrect but plausible. There is a 60 second time delay input into the SI circuit that prevents resetting until after the time delay. This does not components after they have started.

Technical Reference(s): 1-NHY-509048

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 012 Reactor Protection |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.7 |
| Learning Objective: | L8056I23RO |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 2 | |
| Question 39 | Group # | 1 | |
| | K/A # | 012 Reactor Protection A2 Ability to a) predict the impacts of the following malfunctions or operations on the RPS, and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.02 Loss of instrument power | |
| | Importance Rating | 3.6 | |

Proposed Question:
Given the following plant conditions:

- A plant startup is in progress.
- The plant is at 6% power.
- Pressurizer pressure channel PT-456 has failed low.
- The crew has tripped all the required bistables in accordance with OS1201.06, 'Pressurizer Pressure Instrument PT-455/458 Failure'.
- Power is subsequently lost to EDE-PP-1D.

What is the status of the plant?

A. Tripped due to High Pressurizer Pressure or OPΔT.

B. The plant will remain at power. Bistables for PT-458 are tripped.

C. Tripped due to High Pressurizer Pressure or OTΔT, SI actuated.

D. Tripped due to High Pressurizer Pressure or Low Pressurizer Pressure, SI actuated.

Proposed Answer: C

A is incorrect but plausible. A High Pressurizer Pressure coincidence will be met however the pressure instruments feed into the OTΔT circuit versus the OPΔT circuit. It is a common operator misconception regarding which ΔT trip that the pressure instruments feed into.

B is incorrect but plausible. It is plausible that the coincidence for the high pressure trip have not been met as the first instrument failed low, however the process of tripping bistables per OS1201.06 would have place the high pressure trip in a ¼ coincidence. The additional bistable trip due to the loss of PP-1D would have caused a trip.

C is correct. When PT-456 failed the procedure would have placed High Pressurizer Pressure, OTΔT, Low Pressurizer Pressure Trip, and SI P-11 in a tripped condition. The loss of PP-1D would have resulted in meeting the coincidence for the High Pressurizer Pressure Trip, OTΔT Trip and SI.

D is incorrect but plausible. It is plausible that the Low Pressurizer Pressure trip would have actuated however power level is less than the P-7 interlock (10% power). The Low Pressurizer Pressure Trip is not active below P-7.

Technical Reference(s): 1-NHY-509046

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 012 Reactor Protection |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/43.5/45.3/45.5 |
| Learning Objective: | L8056I17 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 40 | Group # | 1 | |
| | K/A # | 013 Engineered Safety Features Actuation<br>A4 Ability to manually operate and/or monitor in the control room:<br>A4.01 ESFAS-Initiation Equipment Which Fails to Actuate | |
| | Importance Rating | 4.5 | |

Proposed Question:
The crew has tripped the reactor due to a steamline break.

Given the following plant conditions:

- Safety Injection is actuated.
- Steam Generator 'A' Pressure is 800 psig and STABLE.
- Steam Generator 'B' Pressure is 760 psig and STABLE.
- Steam Generator 'C' Pressure is 800 psig and STABLE.
- Steam Generator 'D' Pressure is 800 psig and STABLE.
- Reactor Coolant System pressure is 1880 psig and DECREASING.
- Containment Pressure is 6.1 psig and INCREASING.

Which of the following describes ALL of the additional ESFAS actuations that should have occurred?

A. Containment Isolation Phase 'A' ONLY.

B. Containment Isolation Phase 'A' and Main Steam Isolation ONLY.

C. Containment Isolation Phase 'A', Main Steam Isolation, and Containment Spray ONLY.

D. Containment Isolation Phase 'A', Main Steam Isolation, Containment Spray, and Containment Isolation Phase 'B'.

Proposed Answer: B

A is incorrect but plausible. There would be a Phase 'A' signal however there would also be a The Main Steamline Isolation based on Containment Pressure >4.3 psig. The answer is plausible because the Main Steam Isolation setpoint based on SG pressure is 585 psig which is not met. The student could believe that there was no need for a Main Steam Isolation.

B is correct. The Phase 'A' Isolation would occur due to the SI signal. The Main Steamline Isolation would be based on Containment Pressure >4.3 psig.

C is incorrect but plausible. A High Containment Pressure signal (>4.3 psig) does exist, however the CBS actuation does not occur until containment pressure is >18 psig.

D is incorrect but plausible. A High Containment Pressure signal (>4.3 psig) does exist, however the CBS actuation does not occur until containment pressure is >18 psig. Additionally, the Phase 'B' isolation signal occurs based on a CBS actuation.

| | | |
|---|---|---|
| Technical Reference(s): | 1-NHY-509047, FW System Gen Trip Signals | 1-NHY-509048, Safeguard Actuation Signals |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 013 Engineered Safety Features Actuation |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.5 to 45.8 |
| Learning Objective: | L8057I08 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 41 | Group # | 1 | |
| | K/A # | 022 Containment Cooling K2 Knowledge of power supplies to: K2.01 Containment Cooling fans. | |
| | Importance Rating | 3.0 | |

Proposed Question:
The plant is initially operating at 100% power. Containment Structure Cooling fans 1A, 1B, 1C, 1D, and 1F are in service.

Given the following conditions:

- Safety Injection Occurs.
- Containment pressure increases to 16 PSIG and then slowly decreases.
- A loss of offsite power occurs.
- Bus E-5 power is restored by Emergency Diesel Generator 'A'
- The Train 'A' Emergency Power Sequencer has completed its step sequence.
- Bus E-6 failed to re-energize.

What is the status of the Containment Structure Cooling Fans?

A. No fans are running.

B. Fans 1C and 1F are running.

C. Fans 1A, 1B, and 1D are running.

D. Fans 1A and 1C are running.

Proposed Answer: A

A is correct. The fans will trip on Loss of Power. Since SI is not reset the fans will not start off of the sequencer.

B is incorrect but plausible. Fans 1C and 1F are powered from bus 5 however SI is not reset the fans will not start off of the sequencer.

C is incorrect but plausible. Fans 1A, 1B, and 1D are powered from Bus 6. The students may confuse these fans power supplies as being from Bus 5. The students would choose this answer based on a common misconception of the fan power supplies.

D is incorrect but plausible. The labeling sequence of 1A/1C is in alignment with the conventional Train A labeling standard for electrical equipment. The students would choose this answer based on a common misconception of the fan power supplies.

| | |
|---|---|
| Technical Reference(s): | 1-NHY-310931 Sheet AC5b/L803b |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 022 Containment Cooling |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.6 |
| Learning Objective: | L8020I08, L8020I09, L8038I04 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 42 | Group # | 1 | |
| | K/A # | 026 Containment Spray K4 Knowledge of the CSS design features and/or interlocks which provide for the following: K408 Automatic swapover to containment sump suction for recirculation phase after LOCA (RWST low-low level alarm). | |
| | Importance Rating | 4.1 | |

Proposed Question:
Given the following plant conditions:

- A Large Break LOCA has occurred.
- The crew has entered E-1, 'Loss of Reactor or Secondary Coolant'.
- Safety Injection has been RESET.
- VAS Alarm 'RWST Level Lo-Lo' is in ALARM.
- RWST level is 120,000 gallons and decreasing.

Which of the following describes the response of CBS-V-8 and CBS-V-14 (Containment Recirc Sump Suction Valves)?

A. RWST level has reached the semi-automatic swapover setpoint. The valves will not open because the SI signal has been RESET.

B. RWST level has not reached the semi-automatic swapover setpoint. The crew should continue in E-1 until the setpoint is reached.

C. RWST level has not reached the semi-automatic swapover setpoint. The valves must be manually opened per ES-1.3, 'Cold Leg Recirculation'.

D. RWST level has reached the semi-automatic swapover setpoint. The valves will open because the SI signal is still present to the CBS suction swapover circuit.

Proposed Answer: D

A is incorrect but plausible. The swapover setpoint has been reached. While it is true that the Safety Injection signal has been reset the valves will still reposition because the SI signal that feeds into the swapover circuit still exists.

B is incorrect but plausible. The RWST swapover setpoint is at 120,478 gallons.

C is incorrect but plausible. The swapover setpoint has been reached. There is no need to manually align the valves per ES-1.3.

D is correct. CBS-V-8/14 will open with RWST level Lo-Lo (120478 gallons) coincident with SI input to swapover ciruit.

| | | |
|---|---|---|
| Technical Reference(s): | 1-NHY-503258, CBS-ECCS/Spray Recirc Signal Generation | 1-NHY-503252, CBS-Cont. Sump Valves Logic Daigram |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 026 Containment Spray |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7 |
| Learning Objective: | L8035I03, L8035I13 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 2 | |
| Question 43 | Group # | 1 | |
| | K/A # | 039 Main and Reheat Steam<br>K5 Knowledge of the operational implications of the following concepts as they apply to the MRSS:<br>K5.05 Basis for RCS cooldown limits. | |
| | Importance Rating | 2.7 | |

Proposed Question:
The following plant conditions exist:

- The plant is in Mode 3.
- RCS Temperature is 375°F.
- 'C' RCP is operating.
- Cool down is in progress with MS-PK-507 in AUTO in Steam Pressure Mode.
- OS1000.04 'Plant Cooldown From Hot Standby to Cold Shutdown' limits the cool down rate to 100°F/Hr.

What is the basis for this cool down rate limit?

A. Cool down rate in excess of 100°F/Hr will initiate a crack on the inner wall of the reactor vessel.

B. High cool down rates produce tensile stresses on the vessel inner wall which may exceed allowable stress.

C. High cool down rates produce tensile stresses on the vessel outer wall which may exceed allowable stress.

D. Cool down rate in excess of 100°F/Hr will cause exceeding the 320°F differential temperature limit between the PZR and charging.

Proposed Answer: B

A is incorrect but plausible. Excessive stresses on the inner wall may propagate an existing flaw, not initiate a crack.

B is correct. Basis for cool down rate limit is to prevent the tensile thermal stresses adding to total stress on the inner wall of the Reactor vessel and exceeding allowable $RT_{NDT}$.

C is incorrect but plausible. Basis is tensile stresses on inner wall not the compressive stress on the outer wall.

D is incorrect but plausible. 320°F is limit between PZR vapor space and PZR spray fluid. This limit is to prevent thermal shock to the spray nozzle. With 'C' RCP in operation spray fluid is from RCS not charging.

| | | |
|---|---|---|
| Technical Reference(s): | OS1000.04 | Tech. Spec. Basis 3.4.4.9, Cooldown |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 039 Main and Reheat Steam |
| Question Source: | New |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.5/45.7 |
| Learning Objective: | L1171I01 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 44 | Group # | 1 | |
| | K/A # | 059 Main Feedwater K3 Knowledge of the effect that a loss or malfunction of the MFW will have on the following: K3.02 AFW System | |
| | Importance Rating | 3.6 | |

Proposed Question:
Which one of the following plant conditions will cause an Emergency Feedwater Actuation?

A. Main Steam Isolation Signal

B. P-4 signal combined with Low Tavg

C. 2/2 Main Feedwater Pump Trip Signals

D. 2/4 Level Detectors <20% on any One Steam Generator

Proposed Answer: D

A is incorrect but plausible. A Main Steam isolation signal would isolate steam flow to the main feedwater pumps. It is plausible that the signal could input into the EFW actuation signal as an anticipatory signal for a loss of heat sink. This would be more in line with the AMSAC input into EFW. This answer is also plausible as isolating main steam is an operator action when mitigating an ATWS event.

B is incorrect but plausible. This is a Feedwater Isolation Signal vice an EFW actuation signal.

C is incorrect but plausible. This condition would be anticipatory of a loss of heat sink event, however that event is mitigated by the AMSAC EFW actuation signal.

D is correct. An EFW actuation will occur on a) any SI signal, b) SG narrow range low level (<20%, 2 of 4 on any SG), a Loss of Power, and AMSAC (SG narrow range level <13% for >12 sec on 1 of 1 channel on 3 of 4 SG's)

Technical Reference(s): 1-NHY-509055, Emergency and Startup Feedwater Pumps Functional Diagram

Proposed references to be provided to applicants during examination: None

K/A Topic: 059 Main Feedwater

Question Source: New

| | |
|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.7/45.6 |
| Learning Objective: | L8045I03 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 45 | Group # | 1 | |
| | K/A # | 061 Auxiliary/Emergency Feedwater<br>K2 Knowledge of bus power supplies to the following:<br>K2.02 AFW electrical driven pumps. | |
| | Importance Rating | 3.7 | |

Proposed Question:
Given the following plant conditions:

- The plant is initially at 100% power.
- The 'B' Emergency Diesel Generator is INOPERABLE.
- SEPS is NOT available.
- A loss of off-site power occurs.
- Plant equipment responds as expected.

Which of the following lists the feedwater sources that are available?

A. Turbine Driven EFW Pump ONLY.

B. Motor Driven EFW Pump AND the Startup Feedwater Pump.

C. Turbine Driven EFW Pump AND the Motor Driven EFW Pump.

D. Turbine Driven EFW Pump AND the Startup Feedwater Pump.

Proposed Answer: D

A is incorrect but plausible. The Motor Driven EFW Pump is not available as it is powered from Bus 6, however the Startup Feedwater Pump is available as it is aligned to Bus 5. Bus 5 would be powered from EDG A

B is incorrect but plausible. The Startup Feedwater Pump is available as it is aligned to Bus 5 however the Motor Driven EFW Pump would not be available as it is powered from Bus 6, and the question stem states that SEPS is not available.

C is incorrect but plausible. The Turbine Driven EFW Pump is available however the Motor Driven EFW Pump would not be available as it is powered from Bus 6, and the question stem states that SEPS is not available.

D is correct. The Turbine Driven EFW Pump is available and the Startup Feedwater Pump is available as it is aligned to bus 5. Bus 5 would be powered from EDG A.

Technical Reference(s): 1-NHY-310844, Sheet A80, 1-P- 1-NHY-310844, Sheet A93, 1-P-

| | |
|---|---|
| 37B | 113 |

1-NHY-310841, Sheets E87/13 and E88/13, MS to EFW Pump Turbine Train A and Train B

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |
| K/A Topic: | 061 Auxiliary/Emergency Feedwater |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7 |
| Learning Objective: | L8045I01 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 46 | Group # | 1 | |
| | K/A # | 061 Auxiliary/Emergency Feedwater<br>K4 Knowledge of the AFW design features and/or interlocks which provide for the following:<br>K408 AFW recirculation. | |
| | Importance Rating | 2.7 | |

Proposed Question:
Given the following plant conditions:

- The plant has tripped from 100% power.
- FW-P-37A 'Steam Driven EFW Pump' has started automatically.
- Procedure E-0, 'Reactor Trip or Safety Injection' immediate actions are complete.
- No other operator actions have been performed.

What is the status of FW-V-346, the FW-P-37A Recirculation Valve and why?

A. Closed to protect the pump from runout flow conditions.

B. Open to the pump suction assuring adequate minimum flow protection.

C. Open to the Condensate Storage Tank assuring adequate minimum flow protection.

D. Closed to ensure minimum design flow rate is available to the intact Steam Generators.

Proposed Answer: D

A is incorrect but plausible. The valve is closed however the design basis is to ensure adequate flow to the intact steam generators.

B is incorrect but plausible. The recirculation valve does provide miniflow protection to the pump when EFW flow is procedurally throttled. There have been no operator actions, so the valve is closed. Additionally, the recirc flowpath is back to the CST not the pump suction.

C is incorrect but plausible. The recirculation valve does provide miniflow protection to the pump when EFW flow is procedurally throttled. There have been no operator actions, so the valve is closed.

D is correct. The valve is closed and would only be opened if operators took procedural action to throttle EFW flow.

Technical Reference(s): 1-NHY-310844, Sheet C3S, 1- 1-NHY-310844, Sheet C3T, 1-

FW-V-346 FW-V-347

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |
| K/A Topic: | 061 Auxiliary/Emergency Feedwater |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.7 |
| Learning Objective: | L8045I05 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 47 | Group # | 1 | |
| | K/A # | 062 AC Electrical Distribution A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls, including: A1.01 Significance of DG load limits. | |
| | Importance Rating | 3.4 | |

Proposed Question:
Given the following conditions:

- Emergency Diesel Generator 'A' is running loaded for a scheduled surveillance.
- An NSO calls the control room and reports that DG load is 6550KW.

What load adjustment would the BOP Operator have to make in order to adhere to the Emergency Diesel Generator continuous load rating limitation?

A. Reduce load by at least 467 KW.

B. Reduce load by at least 47 KW.

C. No load reduction is needed. Load cannot be raised by more than 147 KW.

D. No load reduction is needed. Load cannot be raised by more than 749 KW.

Proposed Answer: A

A is correct. The continuous load rating is 6083. this would required a load reduction of at least 467 KW.

B is incorrect but plausible. This is the load reduction that would bring load below the 2000 hr/ yr rating of 6503KW.

C is incorrect but plausible. This answer lists the load margin from the 168 hr/yr rating.

D is incorrect but plausible. This answer lists the load margin from the 20 min/yr rating.

Technical Reference(s): OS1026.01. Precaution 3.4.

Proposed references to be provided to applicants during examination: None

K/A Topic: 062 AC Electrical Distribution

| | |
|---|---|
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/45.5 |
| Learning Objective: | L8019I02 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 48 | Group # | 1 | |
| | K/A # | 062 AC Electrical Distribution<br>A3 Ability to monitor automatic operation of the electrical distribution system, including:<br>A3.05 Safety related indicators and controls. | |
| | Importance Rating | 3.5 | |

Proposed Question:
Given the following condition:

- The plant is at 100% power.
- Emergency Diesel Generator 1A has been placed in LOCAL control.

How will Emergency Diesel Generator 1A respond to a Loss of Offsite Power?

A. DG 1A must be manually started.
The output breaker will automatically close.
Load sequencing will automatically occur.

B. DG 1A must be manually started.
The output breaker must be manually closed.
Load sequencing must be performed manually.

C. DG 1A will start automatically.
The output breaker must be manually closed.
Load sequencing must be performed manually.

D. DG 1A will start automatically.
The output breaker must be manually closed.
Load sequencing will automatically occur upon breaker closure.

Proposed Answer: D

A is incorrect but plausible. It is a common misconception that the diesel output breaker will auto close on an LOP in the LOCAL configuration. The diesel will start automatically on an LOP signal with the engine in LOCAL.

B is incorrect but plausible. The diesel will still start on an LOP even if in LOCAL. It is a common misconception with regard to the machines status in LOCAL or MAINTENANCE. If the diesel is in MAINTENANCE it will not start.

C is incorrect but plausible. Once the Emergency Power Sequencer receives a signal that the diesel breaker is closed it will start sequencing.

D is correct. If the diesel is in LOCAL it will still automatically start however the output breaker must be manually closed. The EPS will automatically sequence when the output breaker is closed.

| | | |
|---|---|---|
| Technical Reference(s): | 1-NHY-310857, Sheet E93/8, EDG 1A Start Circuit | 1-NHY-310102, Sheet A54, Bus E5 DG 1A Inc. |
| | 1-NHY-310108, Sheet 5a, EPS Logic Diagram | |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 062 AC Electrical Distribution |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.7/45.5 |
| Learning Objective: | L8020I13, L8020I16 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 49 | Group # | 1 | |
| | K/A # | 063 DC Electrical Distribution<br>K1 Knowledge of the physical connections and/or cause-effect relationships between the DC electrical system and the following systems:<br>K1.02 AC electrical system. | |
| | Importance Rating | 2.7 | |

Proposed Question:
Given the following plant conditions:

- A loss of offsite power has occurred.
- The 'A' Emergency Diesel Generator is powering Bus E5.
- The 'B' Emergency Diesel Generator has failed to start.

Which of the following describes the expected electrical power flowpath to vital power panels PP-1A and PP-1B?

A. Bus E51→MCC E512→UPS-I-1A→PP-1A
Battery B-1B→DC Bus 11B→UPS-I-1B→PP-1B

B. Battery Charger BC-1A →DC Bus 11A→UPS-I-1A→PP-1A
Bus E61→MCC E612→UPS-I-1B→PP-1B

C. Battery B-1A→DC Bus 11A→UPS-I-1A→PP-1A
Battery Charger BC-1B→DC Bus 11B→UPS-I-1B→PP-1B

D. Battery Charger BC-1A→DC Bus 11A→UPS-I-1A→PP-1A
Battery B-1B→DC Bus 11B→UPS-I-1B→PP-1B

Proposed Answer: A

A is correct. Bus 5 is still energized, so Bus E51 and MCC512 are both energized. MCC-512 would still be supplying ac power into the inverter. The inverter would be supplying PP-1A. Additionally, Bus 6 is deenergized so Battery bank 1B would be supplying power to DC bus 11B. Inverter 1B would be receiving dc supply and inverting it to ac for delivery to PP-1B.

B is incorrect but plausible. UPS-1A would not be supplied with dc power under these conditions.

C is incorrect but plausible. UPS-1A would not be supplied with dc power under these conditions.

D is incorrect but plausible. The Train B lineup is correct, however the I-1A would not be powered from a dc source.

| | |
|---|---|
| Technical Reference(s): | EAC Detailed Systems- Figure 0.1, Instrument Bus One Line Diagram |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 063 DC Electrical Distribution |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.2 to 41.9/45.7 to 45.8 |
| Learning Objective: | L8018I03 |

| Examination Outline Cross-reference: | | RO | SRO |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 2 | |
| Question 50 | Group # | 1 | |
| | K/A # | 064 Emergency Diesel Generator 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. | |
| | Importance Rating | 4.4 | |

Proposed Question:
Given the following plant conditions:

- A Loss of All AC Power (Site Blackout) has occurred.
- The Load Dispatcher reports that he is ready to restore offsite power to the switchyard.

What action will have to be taken to restore power to Bus E5 from the UAT?

A. Reset RMO and close the UAT breaker.

B. Hold RMO Bypass Switch in BYPASS and close the UAT breaker.

C. Place the Bus E5 Synchronizing Switch in the UAT position, reset RMO, and close the UAT breaker.

D. Place the Bus E5 Synchronizing Switch in the UAT position, hold RMO Bypass Switch in BYPASS, and close the UAT breaker.

Proposed Answer: B

A is incorrect but plausible. There is an operator misconception with resetting RMO. RMO is reset to allow for manual operation of plant equipment. In this event RMO would have been initiated on the LOP, however it cannot reset because the EPS did not sequence.

B is correct. RMO Bypass is used in order to restore offsite power when RMO cannot be reset.

C is incorrect but plausible. There is an operator misconception with resetting RMO. RMO is reset to allow for manual operation of plant equipment. In this event RMO would have been initiated on the LOP, however it cannot reset because the EPS did not sequence.

D is incorrect but plausible. Using RMO bypass is correct, however there is no need to place the Synch switch to the UAT position because bus E5 is a dead bus.

| | |
|---|---|
| Technical Reference(s): | 1-NHY-310102, Bus E5 UAT Incoming Line Close Schematic |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 064 Emergency Diesel Generator |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/43.5/45.12/45.13 |
| Learning Objective: | L8067I04 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 51 | Group # | 1 | |
| | K/A # | 073 Process Radiation Monitor<br>A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM system controls including:<br>A1.01 Radiation levels. | |
| | Importance Rating | 3.2 | |

Proposed Question:
Which of the following describes the automatic response of the Waste Gas System to a high process radiation reading at the Waste Gas Compressors?

A. Waste Gas Outlet Radiation Monitor (RM-6504) alarms and WG-FV-1602, Waste Gas Compressor Discharge Flow Control Valve closes.

B. Waste Gas Inlet Radiation Monitor (RM-6503) alarms and WG-FV-1602, Waste Gas Compressor Discharge Flow Control Valve closes.

C. Waste Gas Inlet Radiation Monitor (RM-6503) alarms and WG-FV-1602, Waste Gas Compressor Discharge Flow Control Valve modulates.

D. Waste Gas Outlet Radiation Monitor (RM-6504) alarms and WG-FV-1602, Waste Gas Compressor Discharge Flow Control Valve modulates.

Proposed Answer: A

A is correct as described in OS1252.01, Attachment A.

B is incorrect but plausible. The discharge flow control valve does close however it gets it's signal from RM-6504, the outlet rad monitor.

C is incorrect but plausible. The discharge flow control valve does receive a signal however it closes. Additionally, the high rad signal comes from RM-6504.

D is incorrect but plausible. The discharge flow control valve does receive a signal from RM-6504, however the valve closes.

Technical Reference(s): OS1252.01, Attachment A.

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 073 Process Radiation Monitor |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.5/45.7 |
| Learning Objective: | L8059I06 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 52 | Group # | 1 | |
| | K/A # | 076 Service Water<br>K3 Knowledge of the effect that a loss or malfunction of the SWS will have on the following:<br>K3.01 Closed cooling water. | |
| | Importance Rating | 3.4 | |

Proposed Question:
Given the following conditions:

- Train 'A' Service Water has been transferred to the Cooling Tower using the normal operating procedure.
- A Loss of Offsite Power occurs.
- Bus 5 is re-energized from DG-1A.

How does SW-V-54, SW-P-110A Discharge Valve, respond when power is restored to Bus 5?

A. SW-V-54 starts to close at EPS step 8 AND starts to open 10 seconds after the SW-P-110A breaker closes.

B. SW-V-54 starts closing immediately after the EDG breaker closes. The valve will remain closed until it is manually opened.

C. SW-V-54 starts closing immediately after the EDG breaker closes AND starts to open 10 seconds after the SW-P-110A breaker closes.

D. SW-V-54 remains open preventing SW-P-110A from auto starting after the EDG breaker closes. Manual action is required to re-establish Train A SW cooling.

Proposed Answer: C

A is incorrect but plausible. This answer is correct in that SW-V-54 will start to open 10 seconds after SW-P-110A starts however the valve will start to close immediately after the EDG breaker closes and does not happen on an EPS sequencing step.

B is incorrect but plausible. The valve will start to close when the EDG breaker closes however there are no manual actions required to reopen the valve. The SW system is specifically aligned as described in answer C below. This lineup is done to ensure that SW cooling is automatically reestablished in the event of a loss of power. It is a common misconception that manual actions would be required to re-establish SW on the cooling tower during a loss of power event.

C is correct. When SW is transferred to the Cooling Tower per procedure the ocean SW pumps (41A and 41C) are placed in Pull-to-Lock and the Cooling Tower Pump (110A) is placed in Normal-After-Start. This switch configuration will create a Tower Actuation signal upon Loss of Power. SW-PV-54 will open 10 seconds after SW-P-110A starts.

D is incorrect but plausible. SW-V-54 will close. This allows SW-P-110A to restart pursuant to re-establishing SW cooling from the cooling tower. It is a common misconception that manual actions would be required to re-establish SW on the cooling tower during a loss of power event.

| | | |
|---|---|---|
| Technical Reference(s): | OS1016.05, Service Water Cooling Tower Operation | 1-NHY-301107, AU2, Cooling Tower Pump 110A<br>1-NHY-301107, CP8, Cooling Tower P-110A Discharge Valve |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 076 Service Water |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.6 |
| Learning Objective: | L8037I06 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 53 | Group # | 1 | |
| | K/A # | 078 Instrument Air<br>K1 Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems:<br>K1.02 Service Air | |
| | Importance Rating | 2.7 | |

Proposed Question:

Which of the following conditions will cause Service Air Header Isolation Valves, SA-V-92 and SA-V-93 to close?

A. Loss of Power

B. Reactor Trip P-4 signal

C. Safety Injection signal

D. Trip of either Sierra Air Compressor.

Proposed Answer: A

A is correct. The header isolation valves are fail close valves and will close on loss of control power from MCC-523/623. If a loss of power event occurs the valves will close and will not reopen until manual action is performed to open them.

B is incorrect but plausible. It is plausible that the service air header would be isolated in the event of a reactor trip in order to preserve instrument air pressure for safety related needs. There is no such signal.

C is incorrect but plausible. It is plausible that the service air header would be isolated in the event of a Safety Injection signal in order to preserve instrument air pressure for safety related needs. There is no such signal.

D is incorrect but plausible. It is plausible that the valves would go closed on a loss of the air compressors in order to preserve instrument air pressure. This feature is accomplished by an auto closure of the valves on decreasing instrument air pressure.

Technical Reference(s): 310863, E46/8a, Service Air

Proposed references to be provided to applicants during examination: None

K/A Topic: 078 Instrument Air

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

| | |
|---|---|
| 10 CFR Part 55 Content: | 41.2 to 41.9/45.7 to 45.8 |
| Learning Objective: | L8023I13 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 54 | Group # | 1 | |
| | K/A # | 078 Instrument Air K3 Knowledge of the effects that a loss or malfunction of the IAS will have on the following: K3.02 Systems having pneumatic valves and controls | |
| | Importance Rating | 3.4 | |

Proposed Question:
Given the following plant conditions:

- The Reactor Coolant System is water solid in Mode 5.
- The 'A' Residual Heat Removal pump is providing shutdown cooling.
- Reactor Coolant System temperature is being maintained at 180°F.
- Instrument Air to the 'A' RHR vault is lost.

Assuming no operator action, which of the following describes the effect on Reactor Coolant System temperature?

A. INCREASES because RH-V-606, RHR Heat Exchanger Outlet Valve fails CLOSED and RH-V-618, RHR Heat Exchanger Bypass Flow Control Valve fails OPEN.

B. DECREASES because RH-V-606, RHR Heat Exchanger Outlet Valve fails OPEN and RH-V-618, RHR Heat Exchanger Bypass Flow Control Valve fails CLOSED.

C. DECREASES because RH-V-606, RHR Heat Exchanger Outlet Valve fails OPEN and RH-V-618, RHR Heat Exchanger Bypass Flow Control Valve fails OPEN.

D. INCREASES because RH-V-606, RHR Heat Exchanger Outlet Valve fails CLOSED and RH-V-618, RHR Heat Exchanger Bypass Flow Control Valve fails CLOSED.

Proposed Answer: B

A is incorrect but plausible. Temperature would increase if the valves failed to the position described in the answer.

B is correct. Temperature would decrease because the heat exchanger outlet valve fails open causing more fluid to flow through the heat exchanger.

C is incorrect but plausible. If both valves failed open it is plausible that temperature would decrease due to more flow rate through the system however the bypass valve fails closed.

D is incorrect but plausible. Temperature would increase if both valves failed closed however the heat exchanger outlet valve fails open.

Technical Reference(s): PID-1-RH-B20663

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 078 Instrument Air |
| Question Source: | Bank **2007 NRC Exam** |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7/45.6 |
| Learning Objective: | L1194I04 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 55 | Group # | 1 | |
| | K/A # | 103 Containment<br>K4 Knowledge of the containment design features (s) and/or interlocks which provide for the following:<br>K4.06 Containment isolation system. | |
| | Importance Rating | 3.1 | |

Proposed Question:
Which of the following correctly lists the associated Safeguard Actuation Signals for CVCS system valves?

A. RCP Seal Return Valves (CS-V-167 and 168) close on a 'T' (Phase A) signal.
Letdown Isolation Valves (CS-V-149 and 150) close on a 'T' (Phase A) signal.
Charging Isolation Valves (CS-V-142 and 143) close on an SI signal.

B. RCP Seal Return Valves (CS-V-167 and 168) close on an SI signal.
Letdown Isolation Valves (CS-V-149 and 150) close on a 'T' (Phase A) signal.
Charging Isolation Valves (CS-V-142 and 143) close on an SI signal.

C. RCP Seal Return Valves (CS-V-167 and 168) close on a 'T' (Phase A) signal.
Letdown Isolation Valves (CS-V-149 and 150) close on an SI signal.
Charging Isolation Valves (CS-V-142 and 143) close on a 'T' (Phase A) signal.

D. RCP Seal Return Valves (CS-V-167 and 168) close on an SI signal.
Letdown Isolation Valves (CS-V-149 and 150) close on an SI signal.
Charging Isolation Valves (CS-V-142 and 143) close on a 'T' (Phase A) signal.

Proposed Answer: A

A is correct. CVCS letdown and seal return inside containment and outside containment valves all isolate on a 'T' signal. The charging isolation valves close on an SI signal.

B is incorrect but plausible. It is plausible that the seal return lines would isolate due to the SI signal based on conserving RCS inventory however they are closed by the 'T' signal as part of the Containment Isolation scheme.

C is incorrect but plausible. It is plausible that the letdown isolation valves would close on an SI signal based on conserving RCS inventory however they are close by the 'T' signal as part of the Containment Isolation scheme.

D is incorrect but plausible. It is plausible that the seal return lines would isolate due to the SI signal based on conserving RCS inventory however they are closed by the 'T' signal as part of the Containment Isolation scheme. It is plausible that the letdown isolation valves would close on an SI signal based on conserving RCS inventory however they are close by the 'T' signal as part of the Containment Isolation scheme.

Technical Reference(s): 1-NHY-503337, CS MOV ESF Actuated Valves    1-NHY-503400, CS MOV ESF Actuated Valves

Proposed references to be provided to applicants during examination: None

K/A Topic: 103 Containment

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.7

Learning Objective: L8057I08

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | RO | SRO |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 2 | |
| Question 56 | Group # | 2 | |
| | K/A # | 001 Control Rod Drive K4 Knowledge of the CRDS design features(s) and/or interlocks which provide for the following: K4.03 Rod Control logic | |
| | Importance Rating | 3.5 | |

Proposed Question:
Given the following plant conditions:

- Rod Control is in automatic.
- Reactor Coolant System Tavg is 589°F.
- Turbine and Reactor Power are equal and stable.
- An instrument failure causes Tref to indicate 583°F

What is the initial direction and speed of the control rods?

A. Inward at 48 steps per minute.

B. Inward at 72 steps per minute.

C. Outward at 48 steps per minute.

D. Outward at 72 steps per minute.

Proposed Answer: B

A is incorrect but plausible. The conditions in the stem of the question would result in inward rod movement however the 48 step per minute rod speed is associated with max speed manual rod movement.

B is correct. Rod speed due to temperature error is programmed to progress from 8 to 72 steps per minute as temperature error progresses from 3°F to 5°F. The conditions in the stem of the question create a condition where Tavg is 6°F above Tref. This would result in inward rod movement at 72 steps per minute.

C is incorrect but plausible. If the student made a math error such that Tavg were less than Tref then rod movement demand could be misinterpreted as being outward. Also, rod speed of 48 steps per minute is associated with max speed manual rod movement.

D is incorrect but plausible. The conditions in the stem of the question would result in rod movement at 72 steps per minute however the rod motion would be inward. If the student made a math error such that Tavg were less than Tref then rod movement demand could be misinterpreted as being outward.

| | | |
|---|---|---|
| Technical Reference(s): | 1-NHY-509049, Rod Control and Blocks Functional Diagram | 1-NHY-509056, Turbine Trip/Runback Functional Diagram. |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 001 Control Rod Drive |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7 |
| Learning Objective: | L8031I05 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 57 | Group # | 2 | |
| | K/A # | 011 Pressurizer Level Control<br>A3 Ability to monitor automatic operation of the PZR LCS, including:<br>A303 Charging and Letdown | |
| | Importance Rating | 3.2 | |

Proposed Question:
Given the following plant conditions:

- Pressurizer Level Control is in automatic.
- Level channel inputs are selected to LT-459 primary and LT-460 backup.
- The following sequence of events occurs:
    - Letdown ISOLATES
    - Pressurizer Heaters trip OFF
    - Charging flow reduces to minimum.
    - Actual Pressurizer Level Increases.

Which of the following instrument failures has occurred?

A. Level Channel 460 has failed HIGH

B. Level Channel 459 has failed HIGH

C. Level Channel 460 has failed LOW

D. Level Channel 459 has failed LOW

Proposed Answer: C

A is incorrect but plausible. If channel 460 failed high it would cause a reduction in charging flow if it were selected as the controlling channel however it is selected as the backup channel.

B is incorrect but plausible. If level channel 459 failed high it would cause a reduction in charging flow however it would not cause a trip of the pressurizer heaters.

C is correct. When channel 460 fails low it causes a letdown isolation and will trip the heaters. The letdown isolation will cause pressurizer level to increase as charging is still in service.

D is incorrect but plausible. If level channel 459 failed low letdown would isolate and heaters would trip, however the channel also feeds into the charging flow control circuit, so charging flow would increase.

Technical Reference(s): 1-NHY-509027, Pressurizer Level Control

Proposed references to be provided to applicants during examination: None

K/A Topic: 011 Pressurizer Level Control

Question Source: Bank

Question Cognitive Level: Higher: Comprehension/Analysis

10 CFR Part 55 Content: 41.7/45.5

Learning Objective: L8027I06

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | RO | SRO |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 2 | |
| Question 58 | Group # | 2 | |
| | K/A # | 014 Rod Position Indication<br>A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including:<br>A1.03 PDIL,PPDIL | |
| | Importance Rating | 3.6 | |

Proposed Question:
Given the following plant conditions:

- The crew is in the process of reducing plant load per OS1231.04, 'Rapid Downpower'.
- Plant power is at 65% and decreasing.
- The crew has performed an initial boration to support the downpower.
- Control Rods are in automatic.
- VAS alarm D7761, CTL Rod Bank D Insertion Limit Low is in alarm.
- VAS alarm D7762, CTL Rod Bank D Insertion Limit Lo-Lo is in alarm.

Per the associated VPRO procedural guidance what action should the crew take next?

A. Withdraw control rods because the Axial Flux Difference Limit has been exceeded.

B. Reduce power to less than 50% because the Axial Flux Difference Limit has been exceeded.

C. Trip the reactor because the MODE 1 Shutdown Margin Limit may have been exceeded.

D. Perform a Rapid Boration because the MODE 1 Shutdown Margin Limit may have been exceeded.

Proposed Answer: D

A is incorrect but plausible. Control rods are inserted too deep into the core however the concern and purpose of the associated alarms are due to a challenge to shutdown margin and not AFD.

B is incorrect but plausible. With Control Bank D at the Lo-Lo insertion limit there would be a pronounced deviation of AFD from the target band however alarms are associated with shutdown margin. Power reduction to less than 50% power is a Tech. Spec. action.

C is incorrect but plausible. It is true that the MODE 1 shutdown margin is challenged however the required action per the VPRO procedures is to Rapid Borate versus tripping the reactor.

D is correct. VPRO D7762 directs the crew to go to OS1202.04, Rapid Boration.

Technical Reference(s): VPRO D7762

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 014 Rod Position Indication |
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/45.5 |
| Learning Objective: | L1190I06 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 59 | Group # | 2 | |
| | K/A # | 017 In-Core Temperature Monitor<br>K4 Knowledge of the ITM design features and/or interlocks which provide for the following:<br>K4.01 Input to subcooling monitors. | |
| | Importance Rating | 3.4 | |

Proposed Question:
Which of the following describes the reason that the Reactor Coolant System wide range pressure transmitters are located outside of containment?

A. Reliability is improved because the transmitters are more likely to survive a seismic event.
B. Reliability is improved because a containment electrical penetration is not necessary.
C. Response time is improved because the transmitters are closer to the instrument cabinets in the control room.
D. Accuracy is improved because the transmitters will not be exposed to the adverse containment environment created by a LOCA.

Proposed Answer: D

A is incorrect but plausible. The qualification of instruments associated with accident monitoring is pursuant to NUREG 0588 and Reg. Guide 1.97 which addresses adverse environmental criteria, including elevated temperature, pressure, radiation, and corrosive products, and seismic events. The location of the instrument outside of containment is not pursuant to seismic protection.

B is incorrect but plausible. It is true that there is no electrical penetration associated with the instrument, as its electrical circuitry is downstream of the hydraulic section of the instrument however this is not the reason for it's location outside of containment.

C is incorrect but plausible. It is plausible that the response times would be shorter as the transmitter is electrically positioned closer to the instrument cabinets in the control room however this is not the reason for it's location outside of containment.

D is correct. The wide range transmitter is located outside containment to minimize inaccuracies that may occur from the harsh environment in containment post-LOCA

Technical Reference(s): Lesson SBK LOP L8058I, Accident Monitoring

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 017 In-Core Temperature Monitor |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.7 |
| Learning Objective: | L8058I05 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 60 | Group # | 2 | |
| | K/A # | 035 Steam Generator K5 Knowledge of the operational implications of the following concepts as they apply to the SG's: K5.01 Effect of secondary parameters, pressure, and temperature on reactivity. | |
| | Importance Rating | 3.4 | |

Proposed Question:
Given the following plant conditions:

- The plant is initially at 100% power.
- All control systems are in automatic.
- The #1 Turbine Control Valve fails CLOSED.
- Interlock C-7A is NOT armed.

What is the initial effect of this transient on Reactor Coolant System Tavg and Steam Generator pressures?

A. All four RCS loop Tavg's increase in response to a decrease in heat removal. All Steam Generator pressures increase accordingly.

B. RCS Loop 1 Tavg increases. Steam Generator 'A' pressure increases. The other Loop Tavg's and Steam Generator pressures decrease accordingly.

C. All four RCS loop Tavg's remain stable as steam dumps stabilize reactor power. All Steam Generator pressures increase as loop $\Delta T$'s decrease.

D. All four RCS loop Tavg's increase in response to a decrease in heat removal. All Steam Generator pressures remain stable as steam dumps open to stabilize power.

Proposed Answer: A

A is correct. If the #1 control valve went closed there would be a decrease in heat removal. The control valves are off of a common header so it would not affect a specific RCS loop. Additionally, the total steam flow would decrease so all steam generator pressures are affected. There is no load rejection signal to the steam dumps (C-7A) so the steam dumps do not mitigate the transient.

B is incorrect but plausible. If the control valve closure were specific to the 'A' steam generator then RCS loop 1 would respond differently than the other loops.

C is incorrect but plausible. If the steam dumps responded to the transient then this answer would be accurate. At 100% power the steam dumps are in the Tavg mode and the C-7A arming signal is applicable. The steam dumps would not respond because C-7A is not armed.

D is incorrect but plausible. It is true that all four steam generator pressures would respond however the steam dumps are not armed and do not respond.

Technical Reference(s): 1-NHY-509050MS Dump Control Functional Diagram

Proposed references to be provided to applicants during examination: None

K/A Topic: 035 Steam Generator

Question Source: Bank

Question Cognitive Level: Higher: Comprehension/Analysis

10 CFR Part 55 Content: 41.5/45.7

Learning Objective: L1405I01

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 61 | Group # | 2 | |
| | K/A # | 041 Steam Dump/Turbine Bypass Control<br>A2 Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:<br>A2.02 Steam valve stuck open. | |
| | Importance Rating | 3.6 | |

Proposed Question:
Given the following plant conditions:

- The plant is at 15% power during a plant startup.
- The Steam Dump MODE Selector Switch is in the STEAM PRESSURE MODE
- MS-PK-507 is in automatic.
- Main Steam Header Pressure Instrument PT-507 fails HIGH.

How will the Steam Dumps respond to this failure and what operator action is required per OS1230.01, 'Steam Header Pressure PT-507 Instrument Failure'?

A. The Steam Dumps will CLOSE. The operator must use MS-PK-507 in MANUAL to reopen the Steam Dumps.

B. The Steam Dumps will OPEN. The operator must place either Steam Dump P-12 Interlock Control Switch to OFF to close the steam dumps.

C. The Steam Dumps will OPEN. The operator must take either P-12 Interlock Control Switch to the NORMAL AFTER RESET/NORMAL AFTER BYPASS position to close the Steam Dumps.

D. The Steam Dumps will CLOSE. The operator must take both P-12 Interlock Control Switches to the NORMAL AFTER RESET/NORMAL AFTER BYPASS position to reopen the steam dumps.

Proposed Answer: B

A is incorrect but plausible. With PT-507 failed MS-PK-507 will not function correctly in the Steam Pressure Mode. Per OS1230.01 MS-PK-507 is adjusted in manual however this is done after the operator has closed one of the two P-12 interlock switches.

B is correct. The steam dumps will come full open. This scenario presents a plant challenge as RCS temperature could reduce below the minimum required temperature for criticality. The operator is directed per procedure to go to OFF with either P-12 interlock switch in order to close the steam dumps to mitigate the transient.

C is incorrect but plausible. The Steam Dumps will open however the P-12 switches should be taken to OFF versus NA RESET/NA BYPASS. The dual functions of the P-12 interlock control switches is a common operator misconception.

D is incorrect but plausible. The Steam Dumps will open versus close. Additionally, the P-12 switches should be taken to OFF versus NA RESET/NA BYPASS. The dual functions of the P-12 interlock control switches is a common operator misconception.

| | |
|---|---|
| Technical Reference(s): | OS1230.01, Steam Header Pressure PT-507 Instrument Failure |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 041 Steam Dump/Turbine Bypass Control |
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/43.5/45.3/45.13 |
| Learning Objective: | L1193I08 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 62 | Group # | 2 | |
| | K/A # | 055 Condenser Air Removal<br>K1 Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems:<br>K1.06 PRM system. | |
| | Importance Rating | 2.6 | |

Proposed Question:
Given the following plant conditions:

- The plant is at 100% power.
- RM-6505, Condenser Air Evacuation Discharge Radiation Monitor is in ALARM.

Which of the following describes the significance of this alarm?

A. Radiation level on RM-6505 indicates which Steam Generator has a tube leak.

B. Radiation level on RM-6505 provides an input to the calculation for an approximate value of Primary to Secondary Leak Rate.

C. Radiation level on RM-6505 is used to determine the need for a reactor trip and SI per OS1227.02, 'Steam Generator Tube Leak'.

D. Radiation level on RM-6505 is used to determine which secondary systems need to be isolated per OS1227.02, 'Steam Generator Tube Leak'.

Proposed Answer: B

A is incorrect but plausible. RM-6505 is used for the Radiation Critical Safety Function Status Tree and is indicative of primary to secondary leakage, however it is common to all steam generators.

B is correct. RM-6505 is used for the calculated primary to secondary leak rate and also used in 1227.02 if the value must be calculated manually.

C is incorrect but plausible. The leak rate is used to determine the rate of plant downpower however reactor trip and SI criteria are based on the threshold of maintaining >7% pressurizer level utilizing two charging pumps.

D is incorrect but plausible. The leak rate of change is used to determine the rate of plant downpower however reactor trip and SI criteria are based on the threshold of maintaining >7% pressurizer level utilizing two charging pumps.

| | |
|---|---|
| Technical Reference(s): | OS1227.02, Steam Generator Tube Leak |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 055 Condenser Air Removal |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.2 to 41.9/45.7 to 45.8 |
| Learning Objective: | L1190I02 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | RO | SRO |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 2 | |
| Question 63 | Group # | 2 | |
| | K/A # | 056 Condensate<br>K1 Knowledge of the physical connections and/or cause-effect relationships between the condensate system and the following systems:<br>K1.03 MFW | |
| | Importance Rating | 2.6 | |

Proposed Question:
The following plant conditions exist:

- 100% power.
- All control systems are in automatic.
- CO-P-30A and CO-P-30C are in operation.
- An electrical fault causes 4160 BUS-3 to trip and lockout.
- All equipment is functioning as designed.

What is the expected response of both Main Feedwater pumps?

A. Tripped due to Lo-Lo S/G water level.

B. Tripped due to low suction pressure.

C. Running, speeds decrease to approximately 4400 RPM.

D. Running, maintaining program ΔP across the main feed reg. valves.

Proposed Answer: B

A is incorrect but plausible. Initiating conditions are a loss of condensate pumps and S/G level will decrease. Decreasing S/G level does not trip MFP's.

B is correct. A loss of 4160 BUS-3 trips both operating condensate pumps. The third condensate pump will start but does not provide sufficient flow at 100% power to provide adequate MFP suction pressure. Both MFP's will trip on low suction pressure.

C is incorrect but plausible. On a reactor trip the reactor trip breakers actuate P-4. P-4 inputs to the MFP's speed control on a reactor trip and reduces both pumps to 4400 RPM.

D is incorrect but plausible. The initiating conditions do not directly trip the MFP's. The feed pumps will continue to operate until suction pressure drops to the low suction pressure setpoint.

| | |
|---|---|
| Technical Reference(s): | 1-NHY-503590<br>1-NHY-503591 |
| Proposed references to be provided to applicants during examination: | None |
| K/A Topic: | 056 Condensate |
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.2 to 41.9/45.7 to 45.8 |
| Learning Objective: | L8062I08 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | RO | SRO |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 2 | |
| Question 64 | Group # | 2 | |
| | K/A # | 075 Circulating Water K2 Knowledge of bus power supplies to the following: K2.03Emergency/Essential SWS Pumps | |
| | Importance Rating | 2.6 | |

Proposed Question:
Given the following sequence of events:

- Service Water Pumps P-41A and P-41B are initially running.
- A loss of offsite power to the 345kv switchyard has occurred.
- Bus E5 is de-energized due to a differential relay/lockout.
- The crew has initiated a manual Safety Injection.
- All safety related systems are functioning as designed.
- No other operator actions have been performed.

Which Service Water Pumps will be running 2 minutes after the Safety Injection is initiated?

A. SW-P-41B ONLY

B. SW-P-41D ONLY

C. SW-P-41C AND SW-P-41D

D. SW-P-41B AND SW-P-41D

Proposed Answer: A

A is correct. The running pump will auto start when the Emergency Power Sequencer reaches Step 8. SW-P-41A will not be running because bus E5 was not re-energized.

B is incorrect but plausible. SW-P-41D is powered from the re-energized bus however the previously running pump (41B) would restart. This is a common misconception as there is a 'preferred pump' logic for SW-P-41D if it had been previously running.

C is incorrect. It is plausible that SW-P-41D could be running doe to the 'preffered pump' logic, however it was not previously running. Also SW-P-41C would not be running as it is powered from the de-energized bus E5.

D is incorrect but plausible. It is plausible that the student could choose this answer due to the common misconception associated with the 'preferred pump' logic. SW-P-41B was previously running so it would be the pump that would restart.

| | |
|---|---|
| Technical Reference(s): | 1-NHY-301107, AQ3/AQ4, service Water Pump Breaker Close Schematic |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 075 Circulating Water |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.7 |
| Learning Objective: | L8037I06 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 2 | |
| Question 65 | Group # | 2 | |
| | K/A # | 079 Station Air<br>K4 Knowledge of the SAS design features(s) and/or interlocks which provide for the following:<br>K4.01 Cross-connect with IAS | |
| | Importance Rating | 2.9 | |

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- The 'A' Sierra (lead compressor) trips.
- The 'B' Sierra (lag compressor) fails to start on decreasing air pressure.
- Service and Instrument Air pressures are DECREASING.

Which of the following describes the response of the Service Air isolation valves, SA-V-92 and SA-V-93 to this transient?

A. AUTOMATICALLY CLOSE at 90 psig decreasing.
AUTOMATICALLY OPEN above 93 psig increasing.

B. AUTOMATICALLY CLOSE at 80 psig decreasing.
AUTOMATICALLY OPEN above 83 psig increasing.

C. AUTOMATICALLY CLOSE at 90 psig decreasing.
Resets to allow MANUAL OPENING above 93 psig increasing.

D. AUTOMATICALLY CLOSE at 80 psig decreasing.
Resets to allow MANUAL OPENING above 83 psig increasing.

Proposed Answer: C

A is incorrect but plausible. The auto close setpoint is 90 psig decreasing however there is no automatic open feature.

B is incorrect but plausible. There is an auto close signal however it is at 90 psig.

C is correct. The valves auto close at 90 psig decreasing. The auto close rest is at 93 psig.

D is incorrect but plausible. The auto closure and reset features are true however the setpoint of 80 psig and reset of 83 psig are incorrect.

Technical Reference(s): ON1242.01, Loss of Instrument Air

Proposed references to be provided to applicants during examination: None

K/A Topic: 079 Station Air

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.7

Learning Objective: L8023I06

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 3 | |
| Question 66 | Group # | | |
| | K/A # | 2.1.3 Knowledge of shift or short-term relief turnover practices. | |
| | Importance Rating | 3.7 | |

Proposed Question:
You are the oncoming reactor Operator returning from a two week vacation. When reviewing logs and journal entries for your position, how far back are you required to review in accordance with NAP-402, 'Conduct of Operations, 'Attachment L: Shift Turnover and Relief'?

A. Review entries from the last 24 hours.

B. Review entries from the last 72 hours.

C. Review as far back as necessary to ensure that you are knowledgeable of plant conditions.

D. Review all logs and journal entries since the last time you were assigned to the position.

Proposed Answer: C

A is incorrect but plausible. For instances where an operator has not been off shift for a considerable period of time NAP-402 dictates reviewing info for at least the last 24 hour period.

B is incorrect but plausible. Previous guidance prior to implementation of NAP-402 defined review requirements in terms of days. This answer is a common misconception as it is based on previous guidance.

C is correct. Per NAP-402 "The on-coming operator shall review the associated logbook back to the last shift worked, or at a minimum, for the previous 24 hours, whichever is shorter. For those cases where individuals have been off shift for a considerable period of time, as far back as necessary to ensure that they are knowledgeable of plant conditions."

D is incorrect but plausible. This wording is associated with the requirements if the individual had not been off shift for an extended period of time.

Technical Reference(s): NAP-402, Attachment L, Shift Turnover and Relief, pg. 3 of 6

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.1.3 Knowledge of shift or short-term relief turnover practices.

Question Source: Bank

| | |
|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.10/45.13 |
| Learning Objective: | L1505I06 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| Question 67 | Group # | | |
| | K/A # | 2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc. | |
| | Importance Rating | 3.9 | |

Proposed Question:
The following plant conditions exist:

- The reactor is tripped and RSS is manned.
- EFW-P-37B is stopped.
- EFW-P-37A is running with 1-FW-V-346 (recirculation to CST) open.
- FW-FI-4214 reads 150 gpm.
- FW-FI-4224 reads 150 gpm.
- FW-FI-4234 reads 125 gpm.
- FW-FI-4244 reads 125 gpm.
- Suction pressure on FW-PI-4208 reads 6.0 psig.
- Suction pressure on FW-PI-4209 reads 8.0 psig.
- Recirculation flow on FW-FI-4279 reads 250 gpm.

Using OS1200.02 Attachment C (provided) what is CST level?

A. 220,000 gallons

B. 240,000 gallons

C. 260,000 gallons

D. 305,000 gallons

Proposed Answer: C

A is incorrect. This value results from total flow equal to 800 gpm, an incorrect use of two pumps operating with suction pressure of 6 psig.

B is incorrect. Total flow incorrectly 600 gpm based on one pump operating with 6 psig suction pressure.

C is correct. Total flow is 800 gpm and is calculated based on one pump operating with 6 psig suction pressure.

D is incorrect. Total flow 800 gpm based on one pump running with an incorrect suction pressure of 8 psig.

| | |
|---|---|
| Technical Reference(s): | OS1200.02, Attachment C, CST Level VS EFW Pump suction Pressure |
| Proposed references to be provided to applicants during examination: | OS1200.02, Attachment C, CST Level VS EFW Pump suction Pressure |
| K/A Topic: | 2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc. |
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.10/43.5/45.12 |
| Learning Objective: | L8210I05 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| Question 68 | Group # | | |
| | K/A # | 2.1.26 Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen, and hydrogen). | |
| | Importance Rating | 3.4 | |

Proposed Question:
Given the following conditions:

- You are conducting a pre-job brief for an evolution that requires an NSO to climb on top of the Service Air Receiver located on the 21 ft. level of the Turbine Building.
- The NSO will be positioned approximately 7 feet above the floor.
- The NSO asks if fall protection is required.

According to the NASH Manual how should you respond?

A. Yes. Per the NASH Manual fall protection is required whenever working above floor level.

B. Yes. Per the NASH Manual fall protection is required when working 6 or more feet above the floor.

C. No. Per the NASH Manual fall protection is required when working 10 or more feet above the floor.

D. No. Per the NASH manual fall protection is not required because there is no temporary floor opening.

Proposed Answer: B

A is incorrect but plausible. If there were a temporary floor opening or the individual were less than 6 feet from a roof this answer may apply.

B is correct. Per the NASH Manual, 'Fall protection shall be worn when working 6 feet or more above floor level".

C is incorrect but plausible. The NASH manual does specifically delineate a height requirement for fall protection however it is 6 feet vice 10 feet.

D is incorrect but plausible. It is true that there are no temporary floor openings as the 21 ft level of the turbine building is a solid concrete floor however the specific 6 ft height criteria still applies.

| | |
|---|---|
| Technical Reference(s): | NASH, Chapter 3, Figure 3-1-1,<br>General Safety Rules, section 6,<br>Fall Protection. |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 2.1.26 Knowledge of industrial safety procedures (such as rotating equipment, electrical, high temperature, high pressure, caustic, chlorine, oxygen, and hydrogen). |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.10/45.12 |
| Learning Objective: | L1516I01 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| Question 69 | Group # | | |
| | K/A # | 2.2.2 Ability to manipulate console controls as required to operate the facility between shutdown and designated power levels. | |
| | Importance Rating | 4.6 | |

Proposed Question:
Given the following plant conditions:

- The plant is in MODE 5.
- The pressurizer is water solid in accordance with OS1000.06, 'Pressurizer Bubble Formation'.
- Per OS1000.06, the Unit Supervisor directs the Reactor Operator to reduce RCS pressure from 355 psig to 330 psig.

What action should the Reactor Operator perform to reduce RCS pressure?

A. Throttle open CS-V-185, Auxiliary Spray Valve.

B. De-energize backup group Pressurizer Heaters as required.

C. Throttle open CS-HCV-189, Letdown Flow Control Valve.

D. Throttle open CS-PCV-131, Letdown Backpressure Control Valve.

Proposed Answer: D

A is incorrect but plausible. Normally aux spray would control pressure if there were a bubble in the pressurizer. Under solid plant conditions opening aux spray would introduce more mass flow into the RCS. OS1001.06 prerequisites dictate that aux. spray be closed.

B is incorrect but plausible. De-energizing backup heaters could reduce pressure if a bubble existed in the pressurizer. There is no bubble in the pressurizer under solid plant conditions.

C is incorrect but plausible. Throttling open CS-HCV-189 would increase letdown flow which could reduce pressure, however per OS1001.06, CS-CV-189 is verified open to ensure over pressure protection. Per OS1001.06, step 3.7, "Whenever the plant is water solid and plant pressure is being maintained by the letdown backpressure control valve, both drag valves must be maintained open".

D is correct. During solid plant operation the letdown backpressure control valve is adjusted as necessary to control RCS pressure.

| | |
|---|---|
| Technical Reference(s): | OS1001.06, Pressurizer Bubble Formation |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 2.2.2 Ability to manipulate console controls as required to operate the facility between shutdown and designated power levels. |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.6/41.7/45.2 |
| Learning Objective: | L1166I06 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| Question 70 | Group # | | |
| | K/A # | 2.2.22 Knowledge of limiting conditions for operation and safety limits. | |
| | Importance Rating | 3.4 | |

Proposed Question:
Given the following plant conditions:

- A Power Operated Relief Valve (PORV) is leaking to the Pressurizer Relief Tank.
- The calculated leak rate is 12.1 gpm.
- All other systems and components are functioning properly.
- No operator action has been taken.

What type of leakage is this and does it exceed the applicable Technical Specification Limiting Condition for Operation?

A. Identified Leakage. Exceeds LCO.

B. Unidentified Leakage. Exceeds LCO.

C. Identified Leakage. Does not exceed LCO.

D. Unidentified Leakage. Does not exceed LCO.

Proposed Answer: A

A is correct. PORV leakage is defined as Identified Leakage. The Tech. Spec. definition for Identified Leakage is "Leakage (except controlled leakage) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank." The Tech. Spec. LCO limit for Identified Leakage is 10 gpm. 12.1 gpm exceeds the LCO.

B is incorrect but plausible. The LCO has been exceeded, however the leakage is Identified Leakage.

C is incorrect but plausible. The leakage is Identified Leakage however the LCO is exceeded.

D is incorrect but plausible. Differentiating between the definition of Identified and Unidentified Leakage has historically been an operator challenge/misconception issue at the facility.

Technical Reference(s): Tech. Spec. 3.4.6.2, Reactor Coolant System Leakage-Operational Leakage

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 2.2.22 Knowledge of limiting conditions for operation and safety limits. |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.5/43.2/45.2 |
| Learning Objective: | L8021115 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| Question 71 | Group # | | |
| | K/A # | 2.3.5 Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable instruments, personel monitoring equipment, etc. | |
| | Importance Rating | 2.9 | |

Proposed Question:
Given the following plant conditions:

- The plant is at 75% power.
- The Steam Generator Blowdown Flash Tank is aligned to the ocean.
- RM-6510, SG A Blowdown Sample Line Radiation Monitor goes into HIGH ALARM.

What automatic action occurs DIRECTLY as a result of this condition?

A. None. This radiation monitor provides no automatic function.

B. SB-CV-6519, Steam Generator Blowdown Flash Tank Outlet Valve will CLOSE.

C. The Steam Generator Blowdown Inside Containment Isolation Valves will CLOSE.

D. The Steam Generator Blowdown Outside Containment Isolation Valves will CLOSE.

Proposed Answer: B

A is incorrect. RM-6510 will close SB-CV-6519. There is an operator misconception as to the operation of the radiation monitors associated with the SB system process. All of the SB sample line radiation monitors will send a CLOSE signal to SB-CV-6519. The misconception is that this signal only comes from the common tank outlet process monitor, RM-6519. which has the same number as the isolating valve itself.

B is correct. Any of the sample line rad. monitors will send a CLOSE signal to SB-CV-6519.

C is incorrect but plausible. The Inside Containment Isolation valves would eventually go closed however this is an indirect effect as they receive a CLOSE signal from a blowdown flash tank high level alarm.

D is incorrect but plausible. The Inside Containment Isolation valves would eventually go closed as an indirect effect from a blowdown flash tank high level alarm however the Outside Containment Isolation Valves do not receive such a signal.

| | |
|---|---|
| Technical Reference(s): | OS1227.01 Recovery From a Steam Generator Blowdown System Isolation. |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 2.3.5 Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable instruments, personel monitoring equipment, etc. |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.11/41.12/43.3/4<br>5.9 |
| Learning Objective: | L8063I03, L8063I16 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 3 | |
| Question 72 | Group # | | |
| | K/A # | 2.3.11 Ability to control radiological releases. | |
| | Importance Rating | 3.8 | |

Proposed Question:
The following plant conditions exist:

- The reactor has tripped and Safety Injection has actuated.
- The crew has diagnosed a rupture on the 'C' Steam Generator.
- The crew is processing E-3, 'Steam Generator Tube Rupture'.
- Step 3 of the procedure directs adjustment of the ruptured Steam Generator ASDV setpoint to 1125 psig.

What is the basis for this action?

A. To maintain at least one SG available for RCS cooldown.

B. To prevent an uncontrolled cooldown of the Reactor Coolant System.

C. To prevent challenging the SG code safety valves and minimize atmospheric radiological release.

D. To increase ruptured SG pressure to the point at which primary-to-secondary leakage will terminate.

Proposed Answer: C

A is incorrect but plausible. The procedure does include a Caution statement that discusses the need to maintain one SG available for cooldown, and the statement is made directly before the step that includes adjusting the ASDV setpoint however the purpose of adjusting the ASDV is to prevent an unisolable radiological release from a code safety valve.

B is incorrect but plausible. If the ASDV were failed open then there could be a cooldown in progress however that is not the intent of the step.

C is correct per the Westinghouse basis document for E-3.

D is incorrect but plausible. E-3 does include a strategy to equalize primary and secondary pressure to stop the leak however that strategy includes cooling down and depressurizing the RCS and does not involve adjusting the ASDV setpoint.

Technical Reference(s): Westinghouse Background Document, E-3.

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 2.3.11 Ability to control radiological releases. |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.11/43.3/45.10 |
| Learning Objective: | L1205I03 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| Question 73 | Group # | | |
| | K/A # | 2.4.6 Knowledge of EOP mitigation strategies. | |
| | Importance Rating | 3.7 | |

Proposed Question:
Given the following plant conditions:

- A loss of secondary heat sink event has occurred.
- The crew has entered FR-H.1, 'Response to Loss of Secondary Heat Sink'.
- Bleed and Feed of the Reactor Coolant System has been established.
- Containment pressure is 2 psig and slowly INCREASING.
- Wide Range Steam Generator Levels are all less than 5% and slowly DECREASING.
- RCS Hot Leg temperatures on all loops are 560°F and INCREASING.
- The crew is about to re-establish EFW flow to the 'D' Steam Generator.

Which of the following describes the action that should be performed to establish feed flow to the 'D' Steam Generator?

A. Feed the Steam Generator at the maximum rate.

B. Do not establish flow until consulting the TSC.

C. Feed at less than or equal to 100 gpm until RCS Hot Leg temperatures are less than 500°F.

D. Feed at less than or equal to 100 gpm until SG level is greater than 14%.

Proposed Answer: A

A is correct. Per FR-H.1 OAS Page a "dry steam generator" is defined as WR level <14% (30% for adverse containment). The OAS CAUTION states 'If bleed and feed has been established AND RCS temperatures are increasing, recovery of a dry SG should be initiated by selecting a single intact SG and feeding at the maximum rate.

B is incorrect but plausible. The procedure states that feed flow to more than one dry SG should not be established until consulting with the TSC.

C is incorrect but plausible. The OAS does discuss feeding at less than or equal to 100 gpm in situations where RCS temperature is stable or decreasing. Additionally, the ability to feed at greater than 100 gpm is not based on temperature criteria but rather SG level.

D is incorrect but plausible. This answer states the technique used if RCS temperature is stable or decreasing.

Technical Reference(s): FR-H.1, Response to Loss of

Secondary Heat Sink

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |
| K/A Topic: | 2.4.6 Knowledge of EOP mitigation strategies. |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.10/45.13 |
| Learning Objective: | L1211I02 |

| Examination Outline Cross-reference: | | | |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | 3 | |
| Question 74 | Group # | | |
| | K/A # | 2.4.11 Knowledge of abnormal condition procedures. | |
| | Importance Rating | 4.0 | |

Proposed Question:
Given the following plant conditions:

- The plant is at 80% power.
- Control Bank 'D' is at 160 steps.
- The Reactor Operator has placed Rod Control in MANUAL to insert rods to control Axial Flux Difference.
- Bank 'D' Control Rods begin to slowly withdraw.

What action is required per OS1210.04, 'Continuous Control Rod Withdrawal'?

A. Place Control Rods in AUTO and verify proper response.

B. Initiate boration as necessary to maintain Tavg within 1.0°F of Tref.

C. Trip the reactor and go to E-0, 'Reactor Trip or Safety Injection'.

D. Place the Rod Control selector switch to the Control Bank 'D' position.

Proposed Answer: C

A is incorrect but plausible. If rods moved outward in manual then it would be plausible to place them in auto to allow the control system to mitigate the temperature transient.

B is incorrect but plausible. With control rods withdrawing, borating is a plausible action to counteract the positive reactivity addition from rod movement outward.

C is correct. Per OS1210.04, if rods are placed in manual and they continue to move then a reactor trip is the proper response.

D is incorrect but plausible. Placing Rod Control to individual bank select is a plausible action to stop the outward rod movement however the procedure calls for a reactor trip.

Technical Reference(s): OS1210.04, Continuous Control Rod Withdrawal

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.4.11 Knowledge of abnormal condition procedures.

Question Source: Bank

| | |
|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.10/43.5/45.13 |
| Learning Objective: | L1184I08 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | 3 | |
| Question 75 | Group # | | |
| | K/A # | 2.4.42 Knowledge of emergency response facilities. | |
| | Importance Rating | 2.6 | |

Proposed Question:
What is the lowest Emergency Plan classification level that requires the Emergency Operations Facility (EOF) to be activated?

A. Unusual Event

B. Unusual Event AND WRGM High Alarm

C. Alert

D. Site Area Emergency

Proposed Answer: C

A is incorrect but plausible. The E Plan is activated by the STED however the EOF is not activated at the UE level.

B is incorrect but plausible. The WRGM does figure into facility activation however it determines the need to activate the Schiller Station facility.

C is correct. Per ER 3.3 the EOF is activated at the Alert level.

D is incorrect but plausible. The EOF is activated at a level higher than UE however it is at the alert level.

Technical Reference(s): ER 3.3, EOF Activation

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.4.42 Knowledge of emergency response facilities.

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.10/45.11

Learning Objective:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| Question 76 | Group # | | 1 |
| | K/A # | 009 Small Break LOCA EA2 Ability to determine or interpret the following as they apply to a small break LOCA: EA2.34 Conditions for throttling or stopping HPI | |
| | Importance Rating | | 4.2 |

Proposed Question:

The crew is performing the actions of E-1, 'Loss of Reactor or Secondary Coolant and is currently evaluating conditions to determine if a transition to ES-1.1, 'SI Termination' is appropriate.

The following conditions exist:

- Containment pressure is 1.5 psig and slowly increasing.
- Subcooling is 57°F and slowly increasing.
- EFW flow is 800 gpm to intact steam generators from the turbine driven EFW pump only.
- Steam Generator narrow range levels are off scale low on all steam generators.
- Steam Generator wide range levels are:
  - 60% in 'A' and 'B' and slowly increasing.
  - 63% in 'C' and 'D' and slowly increasing.
- Pressurizer level is 5% and slowly decreasing.

Which of the following describes the crew's ability to transition to ES-1.1, 'SI Termination' and why?

A. All SI termination conditions are satisfied, the crew may transition to ES-1.1.

B. The crew should NOT transition to ES-1.1. Secondary heat sink is inadequate.

C. The crew should NOT transition to ES-1.1. At least one RCS condition is inadequate.

D. The crew should NOT transition to ES-1.1. Secondary heat sink AND at least one RCS condition are inadequate.

Proposed Answer: C

A is incorrect but plausible. Subcooling and heat sink are satisfactory however pressurizer level is less than the 7% criteria for non-adverse containment conditions.

B is incorrect but plausible. Secondary heat sink is one of the SI termination criteria, however 800 gpm EFW flow satisfies heat sink criteria regardless of adverse containment conditions. Additionally, the distracter is plausible as steam generator level criteria does not meet criteria if containment conditions were adverse. The heat sink criteria is met by having >500 gpm EFW flow regardless of steam generator levels.

C is correct. Pressurizer level is required to be greater than 7% for non-adverse conditions. Pressurizer level criteria is not satisfied.

D is incorrect but plausible. Secondary heat sink is satisfied. 800 gpm EFW flow satisfies heat sink criteria regardless containment conditions. Additionally, the distracter is plausible as steam generator level criteria does not meet criteria if containment conditions were advers. The heat sink criteria is met by having >500 gpm EFW flow regardless of steam generator levels.

| | |
|---|---|
| Technical Reference(s): | E-1, Loss of Reactor or Secondary Coolant. |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 009 Small Break LOCA |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 43.5/45.13 |
| Learning Objective: | L1203I03 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| Question 77 | Group # | | 1 |
| | K/A # | 015 RCP Malfunctions 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of operation. | |
| | Importance Rating | | 4.4 |

Proposed Question:

The following plant conditions exist:

- The plant is at 40% power.
- Alarms for the 'D' Reactor Coolant Pump seal leakoff flow are received.
- 'D' Reactor Coolant Pump total leakoff flow indicates 9.0 gpm and increasing.

Which of the following describes the correct sequence of actions that should be taken in accordance with OS1201.01, 'RCP Malfunction'?

A. Trip the 'D' Reactor Coolant Pump. Take manual control of the 'D' Steam Generator feedwater controller. Close the 'D' Reactor Coolant Pump #1 seal leakoff valve. Commence a plant downpower.

B. Feed the 'D' Steam Generator to 60-70% narrow range level. Trip the 'D' Reactor Coolant Pump. Close the 'D' Reactor Coolant Pump #1 seal leakoff valve.

C. Trip the reactor and enter E-0, 'Reactor Trip or Safety Injection'. After the E-0 immediate actions are complete then stop the 'D' Reactor Coolant Pump and close the #1 seal leakoff valve.

D. Trip the 'D' Reactor Coolant Pump. Close the 'D' Reactor Coolant Pump #1 seal leakoff valve. Trip the reactor and enter E-0, 'Reactor Trip or Safety Injection.

Proposed Answer: B

A is incorrect but plausible. With reactor power less than P-8 (reset 48%) the RCP can be removed from service at power, however, per procedure attachment E the associated steam generator should be fed up to 60-70% level prior to removing the RCP from service.

B is correct. With reactor power less than P-8 (reset 48%) the RCP can be removed from service at power. Per procedure attachment E the associated steam generator should be feed up to 60-70% level prior to removing the RCP from service. Once the RCP is shut down the #1 seal leakoff valve is closed.

C is incorrect but plausible. This would be the correct sequence of actions if reactor power were > P-8.

D is incorrect but plausible. The D RCP is tripped and the #1 seal leakoff valve is closed after the pump is removed from service, however, with reactor power <P-8 there is no need to trip the reactor. Additionally, the 'D' Steam Generator should be fed up to 60-70% level prior to securing the 'D' RCP.

Technical Reference(s): OS1201.01, RCP Malfunction

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 015 RCP Malfunctions |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.10/**43.5**/45.2/45.6 |
| Learning Objective: | L1181I03 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| Question 78 | Group # | | 1 |
| | K/A # | | 022 Loss of Rx Coolant Makeup<br>AA2 Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup:<br>AA2.03 Failures of flow control valves or controllers. |
| | Importance Rating | | 3.6 |

Proposed Question:

Given the following conditions:

- The air supply line has broken off of CS-HCV-182 RCP Seal Flow Control Valve and the valve has repositioned to its "failed position".
- Seal water inlet temperatures to the Reactor Coolant Pumps are:
  - RCP 'A' 164°F
  - RCP 'B' 164°F
  - RCP 'C' 161°F
  - RCP 'D' 163°F

What is the status of seal injection flow and what procedural actions are required for this condition in accordance with OS1202.02, 'Charging System Failure'?

A. Seal injection flow has increased to all RCP's. Based on seal water inlet temperatures the crew should shut down to Hot Standby within 3 hours.

B. Seal injection flow has increased to all RCP's. Based on seal injection flow the crew should monitor VCT level and commence a rapid power decrease.

C. Seal injection flow has decreased to all RCP's. Based on seal water inlet temperatures the crew should trip the reactor, complete the immediate action steps of E-0, 'Reactor Trip or Safety Injection' and then trip the affected RCP's.

D. Seal injection flow has decreased to all RCP's. The crew should make attempts to restore seal injection flow. If flow cannot be restored then the crew should verify one thermal barrier pump running, notify Tech Support, and commence a plant shutdown.

Proposed Answer: D

A is incorrect but plausible. If CS-HCV-182 loses operating air it will fail to the full open position. This will cause seal injection to decrease to minimum. The procedure does have criteria to shut down to Hot Standby based on seal water inlet temperature, however the threshold is >184°F.

B is incorrect but plausible. If CS-HCV-182 loses operating air it will fail to the full open position. This will cause seal injection to decrease to minimum. The procedure does have criteria to monitor VCT level and perform a rapid power decrease, however this action would be performed if VCT level could not be maintained >30%.

C is incorrect but plausible. Seal injection flow would decrease, however the action to trip the reactor would be in the event that seal inlet temperatures exceeded 230°F.

D is correct. CS-HCV-182 is configured such that it works as a backpressure valve for the seal supply lines. The valve fails to the open position. This would cause seal injection flow to decrease to minimum. Procedurally, the crew should attempt to restore seal injection flow. If flow cannot be restored then the crew should verify one thermal barrier pump running, notify Tech Support, and commence a plant shutdown.

Technical Reference(s): OS1202.02, Charging System Failure.

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 022 Loss of Rx Coolant Makeup |
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 43.5/45.13 |
| Learning Objective: | L1445I11 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| Question 79 | Group # | | 1 |
| | K/A # | | 027 Pressurizer Pressure Control System Malfunction AA2.06 Conditions requiring plant shutdown. |
| | Importance Rating | | 3.9 |

Proposed Question:

Given the following plant conditions:

- The plant is operating at 45% power
- Pressurizer Spray Valve RC-PCV-455A is stuck OPEN
- The crew entered OS1201.06, 'Pressurizer Pressure Instrument/Component Failure'.
- All efforts to close the spray valve have failed.
- Pressurizer pressure is at 2000 psig and decreasing steadily.

Which of the following actions should be taken?

A. Trip the 'C' Reactor Coolant Pump and then trip the reactor.

B. Trip the reactor and then trip the 'A' Reactor Coolant Pump.

C. Trip the reactor and then trip the 'C' Reactor Coolant Pump.

D. Feed the 'C' Steam Generator to 60-70% and then trip the 'C' Reactor Coolant Pump.

Proposed Answer: C

A is incorrect but plausible. The 'C' Reactor Coolant Pump is associated with RC-PCV-455A, however, OS1201.06 directs tripping the reactor prior to removing the associated RCP from service.

B is incorrect but plausible. The procedure does direct tripping the reactor and then tripping the associated RCP, however, the 'C' RCP is the pump associated with RC-PCV-455A.

C is correct. The 'C' Reactor Coolant Pump is associated with RC-PCV-455A. OS1201.06 directs tripping the reactor prior to removing the associated RCP from service.

D is incorrect but plausible. The 'C' Reactor Coolant Pump is associated with RC-PCV-455A, however, OS1201.06 directs tripping the reactor prior to removing the associated RCP from service. The procedure does not have guidance for feeding up a stem generator and removing the RCP from service at power.

| | | |
|---|---|---|
| Technical Reference(s): | OS1201.06, PZR Pressure Instrument/Component Failure. | PID-1-RC-B20840 |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 027 Pressurizer Pressure Control System Malfunction |
| Question Source: | **Bank** 2007 NRC Exam |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | **43.5**/45.13 |
| Learning Objective: | L1182I05 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| Question 80 | Group # | | 1 |
| | K/A # | 038 Steam Generator Tube Rupture<br>EA2 Ability to determine and interpret the following as they apply to a SGTR:<br>EA2.10 Flowpath for charging and letdown flows. | |
| | Importance Rating | | 3.3 |

Proposed Question:
Given the following conditions:

- A tube rupture has occurred on the 'A' Steam Generator.
- The crew has entered E-3 'Steam Generator Tube Rupture.
- The 'A' and 'B' Reactor Coolant Pumps are running.
- Normal charging has been established.
- Normal Letdown has been established.
- Pressurizer level is 23% and slowly decreasing.
- The 'A' Steam Generator Narrow Range Level is 85% and slowly increasing.

What action should the crew take? (Reference material provided)

A. Decrease charging flow. Maintain RCS and Ruptured Steam Generator pressures equal.

B. Increase charging flow. Depressurize the RCS using one pressurizer PORV.

C. Increase charging flow. Depressurize the RCS using normal pressurizer spray.

D. Do not adjust charging flow. Depressurize the RCS using normal pressurizer spray.

Proposed Answer: C

A is incorrect but plausible. If pressurizer level were >75% (65% for adverse containment) then procedural guidance dictates decreasing charging flow only if ruptured SG level is increasing. Maintaining RCS and SG pressures equal would be the required action if SG leve were offscale high.

B is incorrect but plausible. The procedural guidance does dictate increasing charging flow and depressurizing the RCS. The procedure would dictate using a PORV if normal spray was not available and normal letdown were not in service. Normal spray is available as RCP 'A' is in service.

C is correct. Per E-3, step 30, if ruptured SG level is increasing and pressurizer level is <30% then charging flow should be increased and the RCS should be depressurized. Normal spray is available as the 'A' RCP is in service.

D is incorrect but plausible. E-3, step 30 does direct depressurizing the RCS, the only condition that would dictate not adjusting charging would be if pressurizer level were between 30% and 50%. Furthermore, the distracter is plausible because normal pressurizer spray is available as RCP 'A' is in service.

Technical Reference(s): E-3, Steam Generator Tube Rupture

Proposed references to be provided to applicants during examination: E-3, Steam Generator Tube Rupture

| | |
|---|---|
| K/A Topic: | 038 Steam Generator Tube Rupture |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 43.5/45.13 |
| Learning Objective: | L1205I02 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
| --- | --- | --- | --- |
| | Tier # | | 1 |
| Question 81 | Group # | | 1 |
| | K/A # | 065 Loss of Instrument Air 2.1.32 Ability to explain and apply system limits and precautions. | |
| | Importance Rating | | 4.0 |

Proposed Question:

Procedure ON1242.01, 'Loss of Instrument Air' provides guidance for tripping the reactor if instrument air pressure degrades to the PCCW system. What is the basis for the associated procedure CAUTION statement and action?

A. The PCCW containment isolation valves fail closed resulting in loss of cooling to the Reactor Coolant Pumps.

B. Flow to the PCCW heat exchangers fails to the "full cooling" position, potentially causing thermal shock to PCCW loads.

C. Flow to the PCCW heat exchangers fails to the "full bypass" position, potentially causing the PCCW pumps to trip on high temperature.

D. The Reactor Coolant Pump thermal barrier cooling system will isolate, causing a loss of cooling to the Reactor Coolant Pump thermal barriers.

Proposed Answer: A

A is correct. The procedure CAUTION statement reads "The containment PCCW isolation valves FAIL CLOSE on loss of air. A reactor trip is required within 10 minutes on a loss of PCCW cooling to the RCP's. Procedure step 4 directs the operators to check the Containment PCCW Isolation valves open. If the valves are not opened the procedure step RNO directs tripping the reactor within 10 minutes if cooling cannot be restored.

B is incorrect but plausible. The PCCW heat exchanger temperature control valves do fail to the full cooling position. There is a NOTE in the procedure that discusses minimum PCCW temperature of 60°F to prevent brittle fracture to the Thermal Barrier heat exchangers.

C is incorrect but plausible. A high temperature condition is undesirable for the PCCW system and the components it cools. Additionally, the PCCW pumps will trip on a high temperature condition. There is, however no procedure CAUTION statement or associated action for PCCW high temperature conditions or PCCW pump trip conditions.

D is incorrect but plausible. PCCW is supplied to the thermal barrier heat exchangers, however the isolation valves are motor operated valves. Additionally, the distracter is plausible as it implies an impact on the RCP's.

| | | |
|---|---|---|
| Technical Reference(s): | ON1242.01, Loss of Instrument Air | PID-1-CC-B20207<br>PID-1-CC-B20209 |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 065 Loss of Instrument Air |
| Question Source: | Bank. |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.10/**43.5**/45.12 |
| Learning Objective: | L1194I03 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| Question 82 | Group # | | 2 |
| | K/A # | | 005 Inoperable/Stuck Control Rod<br>AA2 Ability to determine and interpret the following as they apply to the Inoperable/Stuck Rod:<br>AA2.03 Required actions if more than one rod is stuck or inoperable. |
| | Importance Rating | | 4.4 |

Proposed Question:
Given the following plant conditions:

- A plant transient has caused a reduction in power from 100% to 80%.
- Two Control Bank D rods have become misaligned 15 steps above the remaining rods in their group.
- I&C investigated and found no electrical control system problems or any problems with the rod stepping mechanisms.
- The crew verifies that D7746, Rod Control Urgent Failure is NOT in alarm.

Which of the following lists all the required actions?

A. Immediately trip the reactor.

B. Be in HOT STANDBY within 6 hours.

C. Determine that the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

D. POWER OPERATION may continue provided that a)within 1 hour the remained of the rods in Bank D are aligned to within +/- 12 steps of the inoperable rods. The THERMAL POWER level shall be restricted pursuant to Tech. Spec. 3.1.3.6. and b) the inoperable rods are restored to OPERABLE status within 72 hours.

Proposed Answer: C

A is incorrect but plausible. There is procedural guidance in OS1210.05, 'Dropped Rod' for tripping the reactor if more than one rod is dropped. There is no such guidance for multiple stuck rods.

D is incorrect but plausible. This would be the correct answer if one rod was inoperable but tripable, and also outside of the +/- 12 step alignment threshold. The question stem asks for "all the required actions". Action a of the Tech. Spec. requires SHUTDOWN MARGIN verification in addition to a 6 hour HOT STANDBY requirement.

C is correct. OS1210.02, 'Failure of Control Rod or Rod Bank to Move' includes the following CAUTION statements:

"Control rods that will not move in manual are inoperable. An immovable rod must be verified to be tripable or assumed to be untripable."

"Tripability may be confirmed by verifying a control system failure, usually electrical in nature or a malfunction associated with the rod stepping mechanism."

Based on this guidance the operator must assume that the rods are untripable. Tech Spec. 3.1.3.1, action a is the applicable action for one or more untripable rods.

B is incorrect but plausible. This would be the correct action of the rods were misaligned but operable such that they could be realigned to within +/- 12 steps.

| | | |
|---|---|---|
| Technical Reference(s): | OS1210.02, Failure of Control Rod or Rod Bank to Move.<br>Tech. Spec. 3.1.3, Movable Control Assemblies Group Height | OS1210.05, Dropped Rod |

Proposed references to be provided to applicants during examination: Tech. Spec. 3.1.3, Movable Control Assemblies Group Height

| | | |
|---|---|---|
| K/A Topic: | 005 Inoperable/Stuck Control Rod | |
| Question Source: | Modified from Bank. | |
| Question Cognitive Level: | Higher: Comprehension/Analysis | |
| 10 CFR Part 55 Content: | 43.5/45.13 | 43.2 |
| Learning Objective: | L8031I24 | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| Question 83 | Group # | | 2 |
| | K/A # | 059 Accidental Liquid RadWaste Release 2.3.11 Ability to control radiation releases. | |
| | Importance Rating | | 4.3 |

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- Waste Test Tank 'B' is being prepared for discharge.
- The Liquid Radwaste Test Tank Discharge Radiation Monitor is found to be inoperable.

Which of the following describes the status of the Waste Test Tank discharge? (Reference material provided)

A. The discharge cannot occur until the Liquid Radwaste Test Tank Discharge Radiation Monitor is returned to operable status.

B. The discharge can occur provided samples are taken once per 24 hours if the specific activity is ≤ .01 microCurie/gram DOSE EQUIVILENT I-131.

C. The discharge can occur provided the flow rate is estimated at least once per 4 hours during the actual release. Pump performance curves may be used to estimate flow.

D. The discharge can occur provided two independent samples are analyzed and two technically qualified personnel independently verify the release rate calculations and discharge line valving.

Proposed Answer: D

A is incorrect but plausible. There is an immediate action requirement to suspend a liquid rad release if the setpoint of the process radiation monitor is less conservative than ODCM Specification C.5.1. This is a known immediate action. The candidate could choose this answer if they do not properly apply ODCM Specification C.5.1. An improper setpoint is different than an inoperable monitor.

B is incorrect but plausible. This distracter is associated with the action for an inoperable Steam Generator Blowdown Flash Tank Drain Radiation Monitor, which is also covered under specification table A.5.1-1.

C is incorrect but plausible. This is the required action if the Liquid Radwaste Test Tank Discharge Radiation Monitor flow rate measuring device becomes inoperable. This action is also covered under specification table A.5.1-1.

D is correct. ODCM Specification C.5.1, Radioactive Effluent Monitoring Instrumentation-Liquids, table A.5.1-1, action 29 states:

With the number of channels OPERABLE less than the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that prior to initiating the release

a. At least two independent samples are analyzed in accordance with Surveillance S.6.1.1, and
b. At least two technically qualified members of the station staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

| | |
|---|---|
| Technical Reference(s): | Offsite Dose Calculation Manual, Specification C.5.1, Radioactive Effluent Monitoring Instrumentation-Liquids |
| Proposed references to be provided to applicants during examination: | Offsite Dose Calculation Manual, Specification C.5.1, Radioactive Effluent Monitoring Instrumentation-Liquids |
| K/A Topic: | 059 Accidental Liquid RadWaste Release |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.11/**43.4**/45.10 |
| Learning Objective: | L1512I05 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 1 |
| Question 84 | Group # | | 2 |
| | K/A # | | W/E03 LOCA Cooldown-Depressurization 2.4.20 Knowledge of operational implications of EOP warnings, cautions, and notes. |
| | Importance Rating | | 4.3 |

Proposed Question:
Given the following conditions:

- A LOCA has occurred.
- The crew is processing ES-1.2, 'Post LOCA Cooldown and Depressurization'.
- The 'A' Charging Pump has been stopped.
- No Reactor Coolant Pumps are running.
- Pressurizer level is 52%.
- Sufficient subcooling exists and the crew has stopped the 'A' Safety Injection Pump.

Immediately after stopping the 'A' Safety Injection pump reactor coolant system pressure briefly decreases and then stabilizes. What action should the crew take?

A. Manually reinitiate Safety Injection.

B. Restart the 'A' Safety Injection pump to restore Reactor Coolant System pressure.

C. Restart the 'A' Centrifugal Charging Pump to restore Reactor Coolant System pressure.

D. Monitor subcooling and pressurizer level to ensure that they stabilize above values requiring manual restart of ECCS pumps.

Proposed Answer: D

A is incorrect but plausible. ES-1.2 contains a note that states "After stopping an ECCS pump, RCS pressure should be allowed to stabilize or increase before stopping another ECCS pump." Furthermore, Step 14, Check If One SI Pump Should Be Stopped', provides guidance for checking subcooling and pressurizer level in order to determine the need for restarting ECCS pumps. In any case the procedure does not state to manually reinitiate Safety Injection.

B is incorrect but plausible. There will be a finite time required for RCS pressure to stabilize when the 'A' Safety Injection pump is secured. Per step 14, the need for ECCS pump restart is based on RCS subcooling and Pressurizer level criteria vice RCS pressure.

C is incorrect but plausible. There will be a finite amount of time required for RCS pressure to stabilize when the 'A' Safety Injection pump is secured. Per step 14, the need for ECCS pump restart is based on RCS subcooling and Pressurizer levl criteria vice RCS pressure.

D is correct. Per the procedural note "After stopping and ECCS pump, RCS pressure should be allowed to stabilize or increase before stopping another ECCS pump." It will take a finite amount of time for RCS pressure to stabilize once the 'A' Safety Injection Pump is stopped. Per step 14, the need for ECCS pump restart is based on RCS subcooling and Pressurizer level.

| | |
|---|---|
| Technical Reference(s): | ES-1.2, Post LOCA Cooldown and Depressurization |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | W/E03 LOCA Cooldown-Depressurization |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.10/**43.5**/45.13 |
| Learning Objective: | L1204I02 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | | RO | SRO |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | | 1 |
| Question 85 | Group # | | 2 |
| | K/A # | W/E09&E10 Natural Circ. EA2 Ability to determine and interpret the following as they apply to the Natural Circulation Operations EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility license and amendments. | |
| | Importance Rating | | 3.8 |

Proposed Question:
ES-0.2, 'Natural Circulation Cooldown' instructs the operators to verify that two CRDM fans are running prior to initiating RCS depressurization during the RCS cooldown.

What action is required if less than 2 CRDM fans are running?

A. Maintain RCS subcooling greater than 50°F.

B. Maintain RCS subcooling 100°F to 130°F

C. Maintain cooldown rate in RCS cold legs between 30°/hr and 50°/hr.

D. Wait 88 hours before initiating a cooldown.

Proposed Answer: B

A is incorrect but plausible. Procedure step 13 dictates maintaining 50°F subcooling if at least two CRDM fans are running.

B is correct. Procedure step 14 dictates maintaining RCS subcooling 100°F to 130°F with less than two CRDM fans running.

C is incorrect but plausible. Procedure step 14 does dictate maintaining cooldown rate in RCS cold legs between 30°/hr and 50°/hr, however this rate is irrelevant of CRDM fan configuration.

D is incorrect but plausible. The procedure contains guidance for waiting 88 hours prior to RCS depressurization after the plant has already been cooled down to less than 200°F

Technical Reference(s): ES-0.2, Natural Circulation Cooldown

Proposed references to be provided to applicants during examination: None

K/A Topic: W/E09&E10 Natural Circ.

| | |
|---|---|
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | **43.5**/45.13 |
| Learning Objective: | L1225I06, L1225I14 |

| Examination Outline Cross-reference: | | RO | SRO |
|---|---|---|---|
| | Level | RO | SRO |
| | Tier # | | 2 |
| Question 86 | Group # | | 1 |
| | K/A # | | 004 Chemical and Volume Control 2.2.22 Knowledge of limiting conditions for operation and safety limits. |
| | Importance Rating | | 4.7 |

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- CS-P-2B, Centrifugal Charging Pump 'B' has been removed from service and Danger Tagged for pump outboard bearing replacement.
- The crew has entered the appropriate Technical Specification action statements.
- Subsequently, an NSO reports that RH-P-8A, Residual Heat Removal Pump 'A' has a large puddle of lube oil below it's inboard pump bearing.
- The Shift Manager declares RH-P-8A INOPERABLE.

What action should be taken?

A. With one of the RHR loops inoperable, immediately initiate corrective action to return the RHR loop to OPERABLE status.

B. Apply Tech. Spec. 3.0.3. Within 1 hour initiate action to place the unit in a MODE in which the specification does not apply.

C. Restore both ECCS subsystems to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

D. Restore CS-P-2B, Train 'B' Centrifugal Charging Pump to OPERABLE status as soon as possible. Subsequently, restore RH-P-8A to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours.

Proposed Answer: B

A is incorrect but plausible. This would be the required action for an inoperable loop in MODE 5.

B is correct. The given conditions place 2 trains of ECCS in an INOPERABLE condition. Tech. Spec. 3.5.2, ECCS Subsystems includes actions for 1 train INOPERABLE. There are no actions for 2 INOPERABLE trains, so Tech. Spec. 3.0.3 applies.

Technical Reference(s): Tech. Spec. 3.5.2, ECCS SubSystems    Tech. Spec. 3.0.3

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |
| K/A Topic: | 004 Chemical and Volume Control |
| Question Source: | Bank **2007 NRC Exam** |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/**43.2**/45.2 |
| Learning Objective: | L8010I13 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| Question 87 | Group # | | 1 |
| | K/A # | 008 Component Cooling Water<br>2.4.11 Knowledge of abnormal condition procedures | |
| | Importance Rating | | 4.2 |

Proposed Question:
Given the following conditions

- The crew notices that the 'A' PCCW Head Tank Level is decreasing.
- Hardwire alarm PCCW Head Tank A Level Low is in alarm.
- 'A' PCCW Head Tank Level is 41% and decreasing.
- The crew has entered procedure OS1212.01, 'PCCW System Malfunction'.

Which of the following describes the required procedural actions to respond to these conditions?

A. Trip the reactor. Perform E-0, 'Reactor Trip or Safety Injection' immediate actions. Trip affected Reactor Coolant Pumps.

B. Locally make up to the head tank. Locate and isolate the leak if possible. Check the 'A' PCCW heat exchanger outlet temperatures 65°F to 75°F.

C. Locally make up to the head tank. Locate and isolate the leak if possible. Isolate PCCW to the Waste Process Building, Spent Fuel Pool Heat Exchanger, and RDMS.

D. Locally make up to the head tank. Locate and isolate the leak if possible. If head tank level decreases to less than 36% then isolate PCCW to the Waste Process Building, Spent Fuel Pool Heat Exchanger, and RDMS.

Proposed Answer: C

A is incorrect but plausible. OS1212.01, step 5g directs tripping the reactor and securing the affected RCP's if head tank level drops below 36% level.

B is incorrect but plausible. OS1212.01, step 5 does direct locally making up to the head tank, locating the leak, and isolating if possible. Direction for checking PCCW heat exchanger outlet temperature in the normal band is included in OS1212.01, however it is associated with actions in response to degraded PCCW cooling conditions.

C is correct. PCCW is designed to auto isolate to the WPB and PCCW rad monitor @ 42% level. Procedural step 5f directs isolating PCCW to the Waste Process Building, Spent Fuel Pool Heat Exchanger, and RDMS.

D is incorrect but plausible. 36% is the setpoint for auto isolation of PCCW to the containment building vice the Waste Process Building and RDMS. Procedural step 5h directs tripping the reactor and associated RCP's if level drops below 36%.

| | |
|---|---|
| Technical Reference(s): | OS1212.01, PCCW System Malfunction |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 008 Component Cooling Water |
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.10/**43.5**/45.3/45.5 |
| Learning Objective: | L1445I02 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| Question 88 | Group # | | 1 |
| | K/A # | 012 Reactor Protection<br>A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:<br>A2.06 Failure of RPS signal to trip the reactor. | |
| | Importance Rating | | 4.7 |

Proposed Question:
Given the following conditions:

- The crew has implemented FR-S.1, 'Response to Nuclear Power Generation/ATWS and is at Step 15, 'Verify Reactor Subcritical'.
- Control rods will not insert in Auto or Manual control.
- Boration flow cannot be established to the Reactor Coolant System.
- Power Range NI channels are fluctuating between 10-15% power.
- Tavg is 600°F and slowly increasing.
- All Steam Generator Narrow Range Levels are 10% and stable.
- Total EFW flow is throttled to 400 gpm.

What procedural actions are required in response to these conditions?

A. Allow the RCS to heat up and transition to E-0, 'Reactor Trip or Safety Injection'.

B. Remain in FR-S.1 and maximize feed flow to cool down and depressurize the RCS until boration flow is established.

C. Transition to FR-C.1, 'Response to Inadequate Core Cooling' to minimize cooldown of the RCS. Return to FR-S.1 when boration flow is established.

D. Allow the RCS to heat up. Perform actions of other Functional Restoration Procedures in effect which do not cooldown the RCS. Return to Step 4 of FR-S.1.

Proposed Answer: D

A is incorrect but plausible. FR-S.1 does direct allowing the RCS to heat up for these conditions, however, a transition to E-0 at this point is not correct. Step 15 RNO directs implementing and applicable FRP's and returning to Step 4 of FR-S.1. A return to Step 4 of FR-S.1 facilitates re-evaluation of reactor trip conditions, at which time a transition to procedure and step in effect would be appropriate.

B is incorrect but plausible. It is correct that FR-S.1 would still be in effect, however the reactor is not subcritical at this point and it is desirable to allow the RCS to heat up in order to introduce negative reactivity. Core cooling is a major concern for an ATWS event, so it is conceivable that actions could be taken to address core cooling concerns, however the introduction of negative reactivity is of higher priority. Depressurizing the RCS is a strategy earlier in FR-S.1 if boration flow is inadequate. In this case the question stem indicates that boration flow cannot be established for an unspecified reason.

C is incorrect but plausible. Core cooling is a major concern for an ATWS event, so it is conceivable that actions could be taken to address core cooling concerns, however the introduction of negative reactivity is of higher priority. FR-S.1, Step 15 RNO does discuss transitioning to other FRP procedures, however it states that they should not cooldown or otherwise add positive reactivity to the core. Returning to step 4 of FR-S.1 is directed by the RNO, not when boration is established.

D is correct. FR-S.1, Step 15 RNO states "Continue to borate. IF boration is NOT available, THEN allow the RCS to heat up. Perform actions of other Functional Restoration Procedures in effect which do not cooldown or otherwise add positive reactivity to the core. Return to Step 4."

| | |
|---|---|
| Technical Reference(s): | FR-S.1, 'Response to Nuclear Power Generation/ATWS |

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |
| K/A Topic: | 012 Reactor Protection |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/**43.5**/45.3/45.5 |
| Learning Objective: | L1200I02 |

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| | Tier # | | 2 |
| Question 89 | Group # | | 1 |
| | K/A # | 064 Emergency Diesel Generator<br>2.2.40 Ability to apply Technical Specifications for a system. | |
| | Importance Rating | | 4.7 |

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- The motor driven Emergency Feedwater Pump has been removed from service for maintenance and is INOPERABLE.
- 30 minutes later the 'B' Charging Pump is declared INOPERABLE.
- An hour later the 'A' Emergency Diesel Generator is declared INOPERABLE.
- The Crew has entered Technical Specification 3.8.1.1, A.C. Sources.
- Per the applicable action statement the crew is demonstrating OPERABILITY of the remaining A.C. power sources.

Which of the following describes the correct interpretation of the ACTION requirements for Tech. Spec. 3.8.1.1, A.C. Sources?

A. The Tech. Spec. specifically requires that the motor driven Emergency Feedwater Pump be OPERABLE. Because the motor driven Emergency Feedwater Pump is INOPERABLE the associated ACTION is not met.

B. The Tech. Spec. requires that all required systems, trains, components and devices that rely on the remaining OPERABLE diesel generator are also OPERABLE. Because the motor driven Emergency Feedwater Pump and the 'B' Charging Pump are INOPERABLE the associated ACTION is not met.

C. The Tech. Spec. specifically requires that both the steam driven and motor driven Emergency Feedwater Pumps be OPERABLE. Because the motor driven Emergency Feedwater Pump is INOPERABLE the associated ACTION is not met.

D. The Tech. Spec. requires that all required systems, trains, components and devices that rely on the INOPERABLE diesel generator are also OPERABLE. There is no impact because the 'B' Charging Pump is associated with the OPERABLE diesel generator.

Proposed Answer: B

A is incorrect but plausible. The Tech Spec requires the steam driven EFW pump to be operable. Per the Tech. Spec. basis, having an OPERABLE steam driven EFW pump ensures a diverse feedwater supply to the steam generators. It is conceivable that the Tech. Spec. could specifically require the motor driven EFW in the case where the 'A' Emergency Diesel in INOPERABLE in order to meet the basis of ensuring feedwater supply to the steam generators.

B is correct. Tech. Spec. 3.8.1.1, in addition to the 1 hour requirement for verifying offsite power sources, also requires:

1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

2. When in MODES 1, 2 or 3, the steam driven emergency feedwater pump is OPERABLE.

C is incorrect but plausible. The Tech Spec requires the steam driven EFW pump to be operable. Per the Tech. Spec. basis, having an OPERABLE steam driven EFW pump ensures a diverse feedwater supply to the steam generators. It is conceivable that the candidate could think that the Tech. Spec. required operability of both EFW pumps to meet the basis of ensuring feedwater supply to the steam generators.

D is incorrect but plausible. The Tech. Spec. requires that all required systems, trains, components and devices that rely on the OPERABLE diesel generator are also OPERABLE. The basis for this action is intended to provide insurance that a loss of offsite power condition does not result in a complete loss of safety function or critical features. During a period when either diesel is inoperable. The candidate could misunderstand this basis and believe that the basis for the action is to prevent further vulnerability of the associated trains safety function or critical features.

Technical Reference(s): Tech. Spec. 3.8.1.1, AC Sources; Tech. Spec. Section B3/4.0, Basis, page B3/4, 8-6.

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 064 Emergency Diesel Generator |
| Question Source: | New |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.10/43.2/**43.5**/45.3 |
| Learning Objective: | L8011I25 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| Question 90 | Group # | | 1 |
| | K/A # | 103 Containment<br>A2 Ability to (a) predict the impacts of the following malfunctions or operations on the containment system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:<br>A2.03 Phase A and B isolation. | |
| | Importance Rating | | 3.8 |

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- The crew receives alarms indicating that the following valves have closed:
  - CS-V-149, 'Letdown Line Containment Isolation'
  - CS-V-167, 'RCP Seal Water Return
- The reactor operator confirms that the board indication for both of these valves is now CLOSED.

Based on these conditions, which procedure should be entered and what would be the plant impact if no operator action were taken?

A. Enter OS1205.01, 'Inadvertent Phase A Containment Isolation'. With no operator action pressurizer level would decrease and cause a trip of all Pressurizer heaters .

B. Enter OS1290.01, 'Response to HELB System Actuation or Malfunction'. With no operator action pressurizer level would decrease and cause a trip of all Pressurizer heaters.

C. Enter OS1205.01, 'Inadvertent Phase A Containment Isolation'. With no operator action pressurizer level would increase and cause a reactor trip on high Pressurizer level.

D. Enter OS1290.01, 'Response to HELB System Actuation or Malfunction'. With no operator action pressurizer level would increase and cause a reactor trip on high Pressurizer level.

Proposed Answer: C

A is incorrect but plausible. CS-V-149 and CS-V-167 both close on a Phase A isolation signal. These symptoms are indicative of an Inadvertent Phase A signal. Entry into procedure OS1205.01 is correct, however, pressurizer level would not decrease because CS-V-149 has closed.

B is incorrect but plausible. CS-V-149 does receive a close signal from the HELB system, however CS-V-167 does not.

C is correct. CS-V-149 and CS-V-167 both receive close signals from containment isolation Phase A. With CS-V-149 closed pressurizer level would increase. As level increased the charging flow rate would reduce, however pressurizer level would continue to increase and cause a reactor trip on high pressurizer level.

D is incorrect but plausible. CS-V-149 does receive a close signal from HELB. Pressurizer level would increase, however CS-V-167 does not receive a signal from HELB.

| | | |
|---|---|---|
| Technical Reference(s): | OS1205.01 Inadvertant Phase A Containment Isolation | OS1290.01 Response to HELB System Actuation or Malfunction |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 103 Containment |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/**43.5**/45.3/45.13 |
| Learning Objective: | L1181I14 |

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| | Tier # | | 2 |
| Question 91 | Group # | | 2 |
| | K/A # | | 002 Reactor Coolant<br>A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:<br>A2.02 Loss of coolant pressure. |
| | Importance Rating | | 4.4 |

Proposed Question:
Given the following conditions:

- The plant is at 100% power
- Pressurizer level is 25% and decreasing.
- Pressurizer pressure is 2150 psig and decreasing.
- Containment pressure is 0.5 psig and slowly increasing.
- All Pressurizer Backup Heaters are on.
- Charging flow is 128 gpm.
- Letdown flow is 0 gpm.

Which of the following procedure flowpaths would be expected based on the given plant conditions?

A. OS1201.02, 'RCS Leak', E-0, 'Reactor Trip or Safety Injection', ES-0.1, 'Reactor Trip Response'.

B. OS1201.02, 'RCS Leak', E-0, 'Reactor Trip or Safety Injection', E-1, 'Loss of Reactor or Secondary Coolant'.

C. OS1227.02, 'Steam Generator Tube Leak', E-0, 'Reactor Trip or Safety Injection', ES-0.1, 'Reactor Trip Response'.

D. OS1227.02, 'Steam Generator Tube Leak', E-0, 'Reactor Trip or Safety Injection', E-3, 'Steam Generator Tube Rupture'.

Proposed Answer: B

A is plausible but incorrect. There is a need for a safety injection signal based upon inability to maintain pressurizer level >7% with maximum charging flow and letdown isolated, therefore ES-0.1 would not be entered.

B is correct. OS1201.02, RCS Leak is entered based on decreasing pressurizer level and increasing containment pressure. Since pressurizer level cannot be maintained >7% OS1201.02 directs a reactor trip, safety injection, and transition to E-0. The given plant conditions would require a transition from E-0 to E-1 based on the transition criteria listed in E-0 step 11, (abnormal containment radiation or containment pressure or containment building level).

C is incorrect but plausible. A steam generator tube leak does impact pressurizer level and pressure, however increasing containment pressure defines that the leak is from the RCS vice through the steam generator tubes. A tube leak would be identified by elevated radiation levels from the condenser off-gas monitor and/or alarms associated with primary to secondary leak rate or leak rate rate of change. SI is required so a transition to ES-0.1 would not be made.

D is incorrect but plausible. A steam generator tube leak does impact pressurizer level and pressure, however increasing containment pressure defines that the leak is from the RCS into containment vice from the RCS to the secondary side of the steam generator.

| | | |
|---|---|---|
| Technical Reference(s): | OS1201.02, 'RCS Leak' | E-0, 'Reactor Trip or Safety Injection' |
| | E-1, 'Loss of Reactor or Secondary Coolant'. | |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 002 Reactor Coolant |
| Question Source: | Modified from bank. |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/**43.5**/45.3/45.5 |
| Learning Objective: | L1180I05, L1202I08, L1203I01, L1203I02 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 2 |
| Question 92 | Group # | | 2 |
| | K/A # | 015 Nuclear Instrumentation A2 Ability to (a) predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.02 Faulty or erratic operation of detectors or compensating components. | |
| | Importance Rating | | 3.5 |

Proposed Question:
Given the following conditions:

- Power range channel N44 was removed from service and it's associated bistables were tripped in accordance with OS1211.04, 'Power Range NI Instrument Failure'.
- Due to a seal problem, the crew has just performed a downpower to 47% and removed the 'D' Reactor Coolant Pump from service.
- 20 minutes later power range channel N41 begins to drift upward and is currently reading 52% power.

What action should be taken?

A. Verify reactor trip and enter E-0, 'Reactor Trip or Safety Injection'.

B. Declare power range channel N41 inoperable and within 1 hour make preparations to be in MODE 3 within the next 6 hours.

C. Bypass channel N41. Coordinate with I&C to troubleshoot channel N41. Place channel N41 associated bistables in the tripped condition within 6 hours.

D. Bypass channel N41. Do not trip channel N41 associated bistables. Immediately initiate corrective actions to return channel N41 or N44 to OPERABLE status.

Proposed Answer: A

A is correct. 2 power range channels are now above the P-8 setpoint (N41 and N44 with a tripped bistable). The reactor should trip on low RCS flow (1 of 4 loops). The reactor should have tripped.

B is incorrect but plausible. The Tech. Spec. actions for NI power range channels only cover the condition for 1 channel inoperable. If two were inoperable the crew should apply Tech. Spec. item 3.0.3. , however with power above P-8 the reactor should have tripped.

C is incorrect but plausible. This distracter describes the directed actions in OS1211.04 (step 2) if channel N41 were the only power range channel that was inoperable, however with power above P-8 the reactor should have tripped.

D is incorrect but plausible. With one channel inoperable step 2 of OS1211.04 directs not tripping additional bistables, as this would cause a reactor trip. There is no Tech. Spec. action that discusses taking immediate corrective actions to return the failed channel to service. This is plausible because Tech. Specs does have a similar statement for multiple failed components, as is the case with the Emergency Feedwater Tech. Spec. Additionally, this distracter is wrong because with power above P-8 the reactor should have tripped.

| | | |
|---|---|---|
| Technical Reference(s): | OS1211.04 Power Range NI Instrument Failure | Tech. Spec. item 3/4. 3.1, Reactor Trip System Instrumentation |

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 015 Nuclear Instrumentation |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/**43.5**/45.3/45.5 |
| Learning Objective: | L1182I10 |

| Examination Outline Cross-reference: | Level | RO | SRO |
| --- | --- | --- | --- |
| | Tier # | | 2 |
| Question 93 | Group # | | 2 |
| | K/A # | 033 Spent Fuel Pool Cooling<br>A2 Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:<br>A2.03, Abnormal spent fuel pool water level or loss of water level. | |
| | Importance Rating | | 3.5 |

Proposed Question:
Given the following conditions:

- The plant is at 100% power.
- The crew has entered OS1215.07, 'Loss of Spent Fuel Pool Cooling or Level' based on computer alarms associated with a loss of spent fuel pool level.
- Spent fuel pool level is 23 feet and decreasing rapidly.
- The leak location has been identified as being in the spent fuel pool

What procedurally directed actions should be taken?

A. Commence makeup from the Chemical and Volume Control System.

B. Commence emergency makeup and close the fuel transfer gate valve.

C. Commence makeup from the Demineralized Water System and stop the spent fuel pool skimmer pump.

D. Commence emergency makeup, stop the spent fuel pool skimmer pumps, stop the spent fuel pool cooling pumps.

Proposed Answer: D

A is incorrect but plausible. Per OS1215.07, 'Loss of Spent Fuel Pool Cooling or Level', if level is less than 25.4 ft and decreasing rapidly the emergency makeup should be utilized (step 1 RNO). Makeup from the Chemical and Volume Control System is an option in the procedure step if level is not decreasing rapidly.

B is incorrect but plausible. Per OS1215.07, 'Loss of Spent Fuel Pool Cooling or Level', if level is less than 25.4 ft and decreasing rapidly the emergency makeup should be utilized (step 1 RNO). Performing emergency makeup is appropriate. The distracter is plausible because the procedure directs closing the spent fuel gate if the leak is determined to be in the cask handling or fuel transfer area (step 4).

C is incorrect but plausible. Step 2 RNO does direct stopping the spent fuel pool skimmer pump. Makeup from the Demineralized Water System is utilized if the level is not decreasing rapidly.

D is correct. Per OS1215.07, step 2, if level is less than 25.4 feet and decreasing rapidly then emergency makeup, per Attachment A should be utilized and the spent fuel skimmer pumps should be stopped. Additionally, the step directs stopping the spent fuel pool cooling pumps if level is less than 23.75 feet.

Technical Reference(s): OS1215.07, Loss of Spent Fuel Pool Cooling or Level

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 033 Spent Fuel Pool Cooling |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.5/**43.5**/45.3/45.13 |
| Learning Objective: | L1192I07 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| Question 94 | Group # | | |
| | K/A # | 2.1.20 Ability to interpret and execute procedure steps. | |
| | Importance Rating | | 4.6 |

Proposed Question:
Which of the following correctly describes the application of a CAUTION statement that preceeds Step 22 of an Emergency Operating Procedure?

A. Applies ONLY to Step 22 "Action/Expected Response" items.

B. Applies to all future "Action/Expected Response" items of the current EOP.

C. Applies to Step 22 "Action/Expected Response" items AND "Response Not Obtained" items.

D. Applies to all future "Action/Expected Response" items AND "Response Not Obtained" items of the current EOP.

Proposed Answer: C

A is incorrect but plausible. While it is correct that a CAUTION that preceeds a step applies to that particular step the CAUTION applies to the whole step, including the RNO item.

B is incorrect but plausible. Some CAUTION statements may apply to the entire EOP procedure, however OP-9.2, EOP Users Guide states "A NOTE or CAUTION which preceeds the <u>first</u> operator action step may apply to the entire procedure.

C is correct. OP-9.2 states "In general, NOTES and CAUTIONS apply to the step which they preceed.

D is incorrect but plausible. OP-9.2, EOP Users Guide states "A NOTE or CAUTION which preceeds the <u>first</u> operator action step may apply to the entire procedure. While it is true that the CAUTION will apply to the Action/Expected action and RNO portion of the applicable step, it does not apply to the remainder of the procedure unless it is stated prior to the first action step in the procedure.

Technical Reference(s): OP-9.2 EOP Users Guide

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.1.20 Ability to interpret and execute procedure steps.

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

| | |
|---|---|
| 10 CFR Part 55 Content: | 41.10/**43.5**/45.12 |
| Learning Objective: | L1195I04 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| Question 95 | Group # | | |
| | K/A # | 2.1.41 Knowledge of refueling process | |
| | Importance Rating | | 3.7 |

Proposed Question:
Given the following plant conditions:

- Refueling is in progress.
- The refueling team is reloading the core.
- The refueling machine interlock override feature must be used to support core packing.

Which of the following requirements must be met before using the refueling machine override feature?

A. The Shift Manager must be present on the refueling machine.

B. The Refueling SRO must act as a spotter and continuously observe the fuel assembly.

C. The Refueling SRO must continuously monitor weight, height, or core location parameters.

D. Each use of the interlock override feature must be distinctly communicated to the Control Room.

Proposed Answer: D

A is incorrect but plausible. The Shift Manager and the Refueling SRO must agree upon the use of the interlock override feature however the SM is not required to be present on the machine. Per OS1015.04 the Refueling SRO directly supervises all actions when the interlock override feature is enabled.

B is incorrect but plausible. A spotter must continuously observe the fuel assembly during interlock override operation however it is not the Refueling SRO who performs this function.

C is incorrect but plausible. Weight, height, and core location must be continuously monitored however it is the Refueling Machine Operator who performs this function.

D is correct. Per OS1015.04, Refueling Machine Operation, Figure 2: Use of Refueling Machine Interlock Override Features EACH use of an interlock override feature is distinctly communicated to the control room.

Technical Reference(s): OS1015.04, Refueling Machine Operation

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 2.1.41 Knowledge of refueling process |
| Question Source: | Bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.2/41.10/**43.6**/45.13 |
| Learning Objective: | L8069I09 |

Seabrook Station 2009 Licensed Operator Remediation Written Exam
ES-401-5 Written Examination Question Worksheet

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| Question 96 | Group # | | |
| | K/A # | 2.2.5 Knowledge of the process for making design or operating changes to the facility. | |
| | Importance Rating | | 3.2 |

Proposed Question:
Which of the following processes is used to determine whether an activity requires an evaluation in accordance with 10CFR50.59?

A. 10CFR50.59 screening

B. 10CFR50.59 applicability determination

C. Equipment Operability Determination

D. SQR (Station Qualified Reviewer) review

Proposed Answer: A

A is correct. A 10CFR50.59 screening is performed to determine if an activity covered by the 10CFR50.59 process requires an evaluation.

B is incorrect but plausible. An applicability determination is part of the 10CFR50.59 administrative process, however the applicability determination is used to see if the activity is covered by 10CFR50.59 or another regulatory program.

C is incorrect but plausible. An operability determination is is used to determine Tech. Spec. operability of plant equipment, which is not part of the 10CFR50.59 process.

D is incorrect but plausible. The procedure change process includes both the 10CFR50.59 screening process and the Station Qualified Reviewer process, however the SQR process is not the part of the process that determines the need for a 10CFR50.59 evaluation.

Technical Reference(s): 10CFR50.59

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.2.5 Knowledge of the process for making design or operating changes to the facility.

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.10/**43.3**/45.13

Learning Objective: L5057I10

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| Question 97 | Group # | | |
| | K/A # | 2.2.38 Knowledge of conditions and limitations in the facility license. | |
| | Importance Rating | | 4.5 |

Proposed Question:
In accordance with Technical Specifications, what is the minimum crew composition during CORE ALTERATIONS?

A. One Shift Manager, One licensed SRO responsible for fuel handling, one licensed RO in the Control Room, and one NSO.

B. One licensed SRO in the Control Room, one licensed SRO responsible for fuel handling, one licensed RO in the Control Room, and two NSO's.

C. One licensed SRO in the Control Room, two licensed RO's in the Control Room, and two NSO's.

D. One Shift Manager, One licensed RO in the Control Room, and one NSO

Proposed Answer: A

A is correct per Tech. Spec. Table 6.2-1

B is incorrect but plausible. There is a requirement for a licensed SRO responsible for fuel handling. There is an NSO requirement, however it is only one. There is a need for an SRO level person, however it is a Shift Manager vice an SRO in the Control Room.

C is incorrect but plausible. There is a requirement for licensed SRO's, however one must be a Shift manager and one must be responsible for fuel handling. Additionally, there is a requirement for one NSO, not two.

D is incorrect but plausible. The crew composition listed does include a portion of the requirement, however an SRO responsible for fuel handling is also needed.

Technical Reference(s): Tech. Specs, Section 6.0-Admin Controls, Section 6.2.2, Station Staff

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.2.38 Knowledge of conditions and limitations in the facility license.

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 41.7/41.10/**43.1**/45.10

Content:
Learning Objective: L8010I15

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| Question 98 | Group # | | |
| | K/A # | 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. | |
| | Importance Rating | | 3.7 |

Proposed Question:
Given the following conditions:

- A General Emergency has been declared due to a LOCA Outside Containment.
- An operator volunteers to make an emergency entry into the penetration area to isolate the leak. Isolating the leak would result in a significant reduction in offsite dose, protecting "large populations".
- The operator has been briefed and is fully aware of the risks involved.

What is the maximum exposure that may authorized for this situation?

A. 4000 mrem TEDE

B. 5000 mrem TEDE

C. 10000 mrem TEDE

D. May exceed 25000 mrem TEDE

Proposed Answer: D

A is incorrect but plausible. 4000 mrem TEDE is the maximum dose allowed prior to needing the Plant Manager's approval for an exposure limit upgrade to 4000-5000 mrem under normal plant conditions, per RP5.1, Figure 5.3, Exposure Limit Upgrades for All Personnel.

B is incorrect but plausible. This is the normal federal limit for TEDE. Additionally, it is listed in ER4.3, Figure 2, Emergency Dose Limits, as the limit for "all activities".

C is incorrect but plausible. 10000R is the previous limit for protecting plant equipment.

D is correct. Per ER4.3, Figure 2, Emergency Dose Limits an individual may be authorized to exceed 25000 mrem TEDE for lifesaving or protection of large populations "only on a voluntary basis to persons fully aware of the risks involved".

Technical Reference(s): ER4.3, Radiation Protection During Emergency Conditions, Figure 2, Emergency Dose Limits

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions. |
| Question Source: | Modified from bank |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 41.12/**43.4**/45.10 |
| Learning Objective: | L1525I13 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| Question 99 | Group # | | |
| | K/A # | 2.4.5 Knowledge of the organization of the operating procedure network for normal, abnormal, and emergency evolutions. | |
| | Importance Rating | | 4.3 |

Proposed Question:
Given the following conditions:

- The plant has tripped.
- Safety Injection has actuated.
- While performing ES-1.2, 'Post LOCA Cooldown and Depressurization' the crew noticed an ORANGE path condition for the CORE COOLING Critical safety Function.
- The crew is now implementing procedure FR-C.2, 'Response to Degraded Core Cooling'.
- The Shift Manager/STA reports that a verified RED path condition now exists for the CORE COOLING <u>and</u> CONTAINMENT Critical Safety Functions.

What action should the Unit Supervisor take?

A. Complete the actions of FR-C.2, 'Response to Degraded Core Cooling' and then transition to FR-C.1, 'Response to Inadequate Core Cooling'.

B. Stop performing FR-C.2, 'Response to Degraded Core Cooling' and immediately transition to FR-C.1, 'Response to Inadequate Core Cooling'.

C. Complete the actions of FR-C.2, 'Response to Degraded Core Cooling' and then transition to FR-Z.1, 'Response to High Containment Pressure'.

D. Stop performing FR-C.2, 'Response to Degraded Core Cooling' and immediately transition to FR-Z.1, 'Response to High Containment Pressure'.

Proposed Answer: B

A is incorrect but plausible. The transition to FR-C.1 is correct as it is the higher of the two RED path conditions, however, EOP rules of usage dictate immediately transitioning to the Highest RED path condition.

B is correct. EOP rules of usage dictate immediately transitioning to the Highest RED path condition. The Core Cooling Critical Safety Function is a higher priority function than the Containment Critical Safety Function.

C is incorrect but plausible. Rules of usage dictate completing functional restoration procedures, however, the rules also direct transitioning to the appropriate procedure upon identification of a RED path condition. Additionally, Core Cooling is a higher priority safety function than Containment.

D is incorrect but plausible. Rules of usage do dictate discontinuing an ORANGE path procedure and immediately proceeding to a RED path procedure, however, Core Cooling is a higher priority safety function than Containment.

Technical Reference(s): OP-9.2, EOP Users Guide

Proposed references to be provided to applicants during examination: None

| | |
|---|---|
| K/A Topic: | 2.4.5 Knowledge of the organization of the operating procedure network for normal, abnormal, and emergency evolutions. |
| Question Source: | Bank |
| Question Cognitive Level: | Higher: Comprehension/Analysis |
| 10 CFR Part 55 Content: | 41.10/**43.5**/45.13 |
| Learning Objective: | L1195I05 |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| | Tier # | | 3 |
| Question 100 | Group # | | |
| | K/A # | 2.4.38 Ability to take actions called for in the facility emergency plan, including support or acting as emergency coordinator if required. | |
| | Importance Rating | | 4.4 |

Proposed Question:
Given the following conditions:

- The station has experienced a Loss of Coolant Accident.
- The Emergency Plan has been activated

Which of the following Short Term Emergency Director (STED) responsibilities may be delegated?

A. Authorization of emergency radiation exposures.

B. Decision and direction to notify offsite authorities.

C. Authorization of the information contained on ER 2.0, "State Notification Fact Sheet'.

D. Directing Security to implement procedure GN1332.00, Security Response to a Declared Radiological Emergency.

Proposed Answer: D

A is incorrect but plausible. Coordination of actions in the field is normally performed from the OSC and involves Health Physics support. It is plausible that Health Physics would have the authority to authorize emergency radiation exposures, however ER1.2, Section 2.0, Responsibilities specifically states that this responsibility cannot be delegated.

B is incorrect but plausible. The Work Control Supervisor makes the initial notifications to offsite authorities, however this process is directed to be performed by the STED.

C is incorrect but plausible. The process of filling out the information on the State Notification Fact Sheet is delegated, usually to the Work Control Supervisor, however authorization of the information is the responsibility of the STED.

D is correct. Directing security to implement GN1332.00, Security Response to a Declared Radiological Emergency is a responsibility that can be delegated. This action is typically performed by the Work Control Supervisor in the process of completing the associated ER1.2 Event Checklists for the applicable emergency plan classification level.

Technical Reference(s): ER1.2, Emergency Plan Activation

Proposed references to be provided to applicants during examination: None

K/A Topic: 2.4.38 Ability to take actions called for in the facility emergency plan, including support or acting as emergency coordinator if required.

Question Source: Bank

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.10/**43.5**/45.13

Learning Objective: L1509I25