| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#1** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1<br>007EK1.02 - Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Shutdown margin | |
| | Importance Rating | 3.4 | |

Proposed Question:

Given the following conditions:

- The plant has been operating at 100% power for 90 days.
- Rod control is in manual.
- Reactor Coolant System Tavg is 588.5°F

If a reactor trip occurred the Shutdown Margin would be ________ than if Tavg were at the programed value for 100% power because power defect would add ________ positive reactivity on the trip.

A. lower, less

B. lower, more

C. higher, less

D. higher, more

Proposed Answer: C

Explanation (Optional):

C is correct. Programmed Tavg for 100% power is 589.1°F. With reactor coolant Tavg at 588.5°F power defect would add less positive reactivity as compared to programmed Tavg.

A is incorrect but plausible. It is true that power defect would add less positive reactivity as compared to programmed Tavg; however Shutdown Margin would be higher.

B is incorrect but plausible. The student would choose this answer if they mistakenly thought that Tavg were higher than programmed Tavg and/or made a conceptual error with regard to change in power defect versus change in Tavg.

| | | | | |
|---|---|---|---|---|
| D is incorrect but plausible. It is true that Shutdown Margin would be higher with Tavg being lower than programmed Tavg; however power defect would add less positive reactivity. | | | | |
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OS1000.10, "Operation At Power", Figure 2: Tavg Program | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | • L1160I05<br>• NUC-GFP-RXT-004, Objectives 11 and 12. | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | X | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Ginna 2012 NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | | | |

| | 55.43 | |
|---|---|---|

Comments:

Parent Question:
Ginna 2012 NRC Exam

O-3.2, Shutdown Margin for an Operating Reactor, states that Tavg must be at programmed Tavg.

If O-3.2 is performed at 100% power with actual Tavg=557F (and control rods in manual), the Shutdown Margin (SDM) on a reactor trip would be:

(1) Than the calculated SDM because the power defect would add (2) positive reactivity on the reactor trip.

(1)(2)

A. higher more
B. higher less
C. lower more
D. lower less

Correct Answer: C

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#2** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000009 (EPE 9) Small Break LOCA / 3<br>009EA1.14 - Ability to operate and monitor the following as they apply to a small break LOCA: Secondary pressure control | |
| | Importance Rating | 3.4 | |

Proposed Question:

Given the following conditions:

- The plant was initially at 100% power.
- A Reactor Coolant System (RCS) leak developed and the crew tripped the reactor and manually actuated Safety Injection.
- A loss of all offsite power occurred when the Main Turbine tripped.

How will Steam Generator pressure be controlled?

A. The Atmospheric Steam Dumps will automatically modulate to maintain RCS temperature at 561°F.

B. The Atmospheric Steam Dumps must be manually operated locally at the valves to maintain RCS temperature at 557°F.

C. The Main Steam Dumps will modulate on the Steam Pressure Controller such that RCS temperature is maintained at 561°F.

D. The Main Steam Dumps will modulate on the Plant Trip Controller to control Steam Generator pressure such that RCS temperature is maintained at 557°F.

Proposed Answer: A

Explanation (Optional):

A is correct. With a loss of offsite power the Main Steam Dumps are not available. In this case the ASDVs will modulate to maintain Steam Generator pressure at 1125 psig, which corresponds to 561°F RCS temperature.

B is incorrect but plausible. It is true that secondary side pressure control will be maintained with the ASDVs, however automatic operation of the ASDVs is still available.

C is incorrect but plausible. If offsite power were available then secondary pressure control would be maintained by the Main Steam Dumps, however they are unavailable as the Circ. Water pumps will have de-energized.

| | | | | | |
|---|---|---|---|---|---|
| D is incorrect but plausible. It offsite power were available then the Main Steam Dumps will modulate on the Plant Trip Controller to control Steam Generator pressure such that RCS temperature is maintained at 557°F | | | | | |
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-509050, MS Dump Control Functional Diagram | | | | |
| Proposed references to be provided to applicants during examination: | | | | None | |
| Learning Objective: | L8047I03, L8047I13 | | | | |
| Question Source: | Bank # | | | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | X | | | |
| Question History: | Last NRC Exam | | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | | |
| | Comprehension or Analysis | X | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | | |
| | 55.43 | | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#3** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000011 (EPE 11) Large Break LOCA / 3<br>011EK3.13 - Knowledge of the reasons for the following responses as the apply to the Large Break LOCA: Hot-leg injection/recirculation | |
| | Importance Rating | 3.8 | |

Proposed Question:

A large break Loss of Coolant (LOCA) occurred approximately 5 hours ago. Why is a transfer to Hot Leg Recirculation initiated?

A. To reduce reactor vessel head temperature

B. To prevent boron precipitation in the reactor core

C. To collapse steam voids in the upper reactor vessel region

D. To remove non-condensable gasses from the reactor vessel

Proposed Answer: B

Explanation (Optional):

B is correct. Per Westinghouse background document ES-1.4, "Transfer to Hot Leg Recirculation", page 1, ECCS is transferred to Hot Leg Recirculation to prevent boron precipitation in the core due to core boiling.

A is incorrect but plausible. Reactor vessel temperature may in fact be elevated, however the primary concern is precipitation of boron within the core/vessel.

C is incorrect but plausible. At this point in the event evolution the upper head region of the vessel may have a steam void, however the primary concern is precipitation of boron within the core/vessel.

D is incorrect but plausible. Non-condensable gasses may accumulate in the vessel head region, however the primary concern is precipitation of boron within the core/vessel.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • Westinghouse Background Document, ES-1.4, "Transfer to Hot Leg Recirculation", pg. 1.<br>• E-1, "Loss of Reactor or Secondary Coolant", step 20. |
| Proposed references to be provided to applicants during examination: | None |

| | | | | |
|---|---|---|---|---|
| Learning Objective: | | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Ginna 2012 NRC Exam | | |

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |

| | | |
|---|---|---|
| 10 CFR Part 55 Content: | 55.41 | **41.5, 41.10** |
| | 55.43 | |
| Comments: | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#4** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000015 (APE 15) Reactor Coolant Pump Malfunctions / 4<br>015AA2.11 - Ability to determine and interpret the following as they apply to<br>the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to jog RCPs during ICC | |
| | Importance Rating | 3.4 | |

Proposed Question:

Given the following plant conditions:

- The crew is implementing FR-C.1, “Inadequate Core Cooling”.
- ECCS flow cannot be restored by any means.
- Steam generator depressurization was ineffective in restoring core cooling.
- All Reactor Coolant Pumps (RCPs) have been stopped.
- All Core Exit Thermocouple temperatures are >/= 1150°F and increasing.

What action is required by FR-C.1 and what is the basis?

A. Start one RCP. The basis is to provide spray flow to depressurize the Reactor Coolant System.

B. Do NOT start any RCPs. Continue attempts to re-establish ECCS flow. The basis is to prevent damage to the RCPs.

C. Do NOT start any RCPs. Continue attempts to depressurize the Steam Generators. The basis is to prevent any additional heat input to the Reactor Coolant System.

D. Start RCPs one at a time in any available idle loop until Core Exit Thermocouples are less than 1100°F. The basis is to force two-phase cooling flow through the core.

| Proposed Answer: | D |
|---|---|

Explanation (Optional):

D is correct. Per FR-C.1, “Response to Inadequate Core Cooling”, step 18, if Core Exit Thermocouple temperatures are greater than 1100°F then the procedure directs checking for an available idle loop and then starting an RCP in that loop. This process continues until CETCs are <1100°F. Per the Westinghouse Background Document, FR-C.1, “Response to Inadequate Core Cooling”, pg. 50, an RCP is started to provide a form of cooling by forcing two-phase cooling flow through the core.

A is incorrect but plausible. It is true that one pump will be started. It is not true that the purpose is to provide spray flow. Providing spray flow is a plausible distractor as the procedure contains guidance for attempting to establish ECCS flow. From a conceptual standpoint, reducing RCS pressure would reduce the discharge head of the ECCS pumps and promote pump flow.

B is incorrect but plausible. FR-C.1 does contain steps for establishing support conditions for starting an RCP. If the student has a misconception regarding the content or intent of those support conditions they may believe that RCS conditions are not met/are detrimental to pump operations.

C is incorrect but plausible. The procedure does contain steps for depressurizing the Steam Generators and it is plausible that the addition of pump heat could be a concern.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • FR-C.1, "Response to Inadequate Core Cooling", step 18.<br>• Westinghouse Background Document, FR-C.1, "Response to Inadequate Core Cooling", pg. 50. |

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L1227I03 | | | |
| Question Source: | Bank # | X | Seabrook TEB#32462 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | n/a | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |

| | | |
|---|---|---|
| 10 CFR Part 55 Content: | 55.41 | |
| | 55.43 | **43.5** |

Comments: none

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#5** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000022 (APE 22) Loss of Reactor Coolant Makeup / 2<br>022AA2.01 - Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: Whether charging line leak exists | |
| | Importance Rating | 3.2 | |

| Proposed Question: | |
|---|---|

Given the following conditions:

- The plant is at 100% power.
- Reactor Coolant System (RCS) Tavg is on program and stable.
- Primary Auxiliary Building (PAB) 7' Area Low Range Radiation Monitor is in Alarm.
- The control room is implementing OS1201.02, "RCS Leak".
- Letdown has been isolated (CS-V-145 and RC-V-81 are closed).
- Charging flow to the RCS loops has been isolated (CS-V-182, CS-V-142, and CS-V-143 are closed).
- Seal Injection flow has been reduced to 8 gpm to each Reactor Coolant Pump.
- Seal return flow is 1.5 gpm per Reactor Coolant Pump.
- Pressurizer level has increased 4.24% over a 10 minute period.
- Volume Control Tank level has decreased 11.15% over a 10 minute period.

What is the source of the leak and what actions are required next?

A. The leak is to the Thermal Barrier System. Go to OS1212.01, "PCCW System Malfunction".

B. The leak is in the Letdown line. Establish excess letdown and then confirm that the leakage has stopped.

C. The leak is in the Reactor Coolant System. Isolate potential RCS leakage sources. If unable to isolate the leak then check for Steam Generator tube leakage.

D. The leak is in the Charging line. Dispatch an NSO to determine the source of the leak. If unable to isolate the leak then go to OS1202.02, "Charging System Failure".

| Proposed Answer: | D | |
|---|---|---|
| Explanation (Optional): | | |

D is correct. OS1201.02, "RCS Leak", step 6 directs the operators to isolate the letdown and charging lines, which is indicated by the conditions listed in the question stem.

Step 7 of the procedure directs the operators to determine if the leak is isolated by calculating whether or not pressurizer level is increasing at a rate equal to the difference between seal injection and seal return. Given the information in the stem of the question and student knowledge that pressurizer level equals 61.3 gallons/%, the calculation is as follows:

Pressurizer level increase : (4.24% X 61.3 gal/%)/10 minutes = 25.99 or 26 gpm

$\Delta$Seal flows: (8 gpm seal inj/pump X 4 pumps)-(1.5 gpm seal return/pump X 4 pumps)=
32 gpm-6gpm=26 gpm

The leak rates are equal, which indicates that the leak is not in the RCS system, but within the boundaries of the CVCS system (either charging or letdown).

Step 8 of OS1201.02 then directs the operators to determine if the VCT level decrease is equal to the difference between seal injection and seal return.

Given the information in the stem of the question and student knowledge that VCT level equals 31 gallons/%, the calculation is as follows:

VCT level decrease: (11.15% X 31 gal/%)/10 minutes = 34.56 gpm

$\Delta$Seal flows=26gpm

The leak rates are not equal, which indicates that there is leakage from the charging portion of CVCS. Step 8 "Response Not Obtained" action directs the operators to dispatch an NSO to determine the source of the leakage, and if the leak is determined to be unisolable, the operators are directed to transition to OS1202.02, "Charging System Failure".

A is incorrect but plausible. It is true that one of the sources of leakage out of the RCS is through the Thermal Barrier Heat Exchanger on any one of the RCPs. It is also true that OS1212.01, "PCCW System Malfunction" provides the guidance for mitigating thermal barrier leaks, however there are no actions in OS1201.02, "RCS Leak" that analyze for thermal barrier leakage or direct operators to transition to OS1212.01, "PCCW System Malfunction".

B is incorrect but plausible. It is true that OS1201.02, "RCS Leak" provides guidance in step 8 and 9 for analyzing for the leak source being in the letdown portion of CVCS, however that would be the case if the VCT level decrease were equal to the difference between Seal Injection flow and Seal Return flow.

C is incorrect but plausible. It is true that OS1201.02, "RCS Leak", step 7 also provides guidance for an analytical determination that the leak is located in the RCS system, however that would be the case if Pressurizer level were increasing at a rate equal to the difference between Seal Injection flow and Seal Return flow.

| | | | |
|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OS1201.02, "RCS Leak" | | |
| Proposed references to be provided to applicants during examination: | None | | |
| Learning Objective: | L1180I05 | | |
| Question Source: | Bank # | x | Seabrook TEB#32266 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | | | |
| | 55.43 | 43.5 | | |

Comments: The analysis in procedure steps 7, 8, and 9 have historically proven to be a conceptual challenge to operators with regard to the intent of what the analysis is proving, i.e., the location of the leak.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#6** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000025 (APE 25) Loss of Residual Heat Removal System / 4<br>025AG2.4.21 - Knowledge of the parameters and logic used to assess the status of safety functions | |
| | Importance Rating | 4.0 | |

Proposed Question:

Which Critical Safety Function status tree(s) take into account Residual Heat Removal pump status?

A. Inventory (I) ONLY

B. Core Cooling (C) ONLY

C. Core Cooling (C) AND Inventory (I)

D. Emergency Recirculation (F) ONLY

Proposed Answer: D

Explanation (Optional):

D is correct. The Emergency Recirculation (F) status tree considers the number of RHR pumps running and also RHR flow is a contributor to the trees "pump runout" calculation.

A is incorrect but plausible. It is plausible that the Inventory status tree would consider RHR pump status as the RHR pumps contribute to inventory recovery during a LOCA.

B is incorrect but plausible. It is plausible that the Core Cooling status tree would consider RHR pump status because the RHR heat exchangers remove heat from the RCS during ECCS injection and recirculation.

C is incorrect but plausible. It is plausible that the Core Cooling status tree would consider RHR pump status because the RHR heat exchangers remove heat from the RCS during ECCS injection and recirculation. Additionally, it is plausible that the Inventory status tree would consider RHR pump status as the RHR pumps contribute to inventory recovery during a LOCA.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • F-0.2, Core Cooling (C) Status Tree<br>• F-0.6, Inventory (I) Status Tree<br>• F-0.7, Emergency Recirculation (F) Status Tree |
| Proposed references to be provided to applicants during | None |

| | | | | |
|---|---|---|---|---|
| examination: | | | | |
| Learning Objective: | L1196I02 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | **43.5** | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#7** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000026 (APE 26) Loss of Component Cooling Water / 8<br>026AG2.4.34 - Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects. | |
| | Importance Rating | 4.2 | |

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- The crew has entered OS1212.01, "PCCW System Malfunction", and is implementing step 5, "Check PCCW System Integrity" and has determined that Train "A" PCCW Head Tank Level is at 41% and slowly decreasing.
- All other plant systems and equipment are operating normally.

What action should be taken next to mitigate the condition, and what plant equipment response will occur if Train "A" Head Tank level continues to decrease?

A. Direct the Primary NSO to perform an emergency fill of the head tank from the Fire Protection system. If level decreases to ≤ 36% then Train "A" PCCW will automatically isolate to Waste Process Building.

B. Direct the Primary NSO to perform an emergency fill of the head tank from the Fire Protection system. If level decreases to ≤ 36% then Train "A" PCCW will automatically isolate to the Containment Building.

C. Direct the Primary NSO to locally make up to the head tank by opening DM-V-13, "DM Water to PCCW Head Tank". If level decreases to ≤ 36% then Train "A" PCCW will automatically isolate to Waste Process Building.

D. Direct the Primary NSO to locally make up to the head tank by opening DM-V-13, "DM Water to PCCW Head Tank". If level decreases to ≤ 36% then Train "A" PCCW will automatically isolate to the Containment Building.

| Proposed Answer: | D |
|---|---|

Explanation (Optional):

D is correct. Step 5 of OS1212.01 directs the crew to perform a local makeup to the Train

"A" PCCW Head Tank from the Demineralized Water System (DM). An automatic isolation of Train "A" PCCW to the Containment Building will occur at 36% level.

A is incorrect but plausible. It is true that OS1212.01 contains guidance for performing an emergency fill from the fire protection system (OS1212.01 Attachment D), however that method utilizes the Fire Protection system vice the RWST, and is only performed when the affected PCCW train is the only one available to cool safety related equipment. The stem of the question states that all plant equipment is operating normally for 100% power, which means that Train "B" PCCW is operating. It is also true that Train "A" PCCW will automatically isolate to the Waste Process Building if level continues to decrease, however the setpoint is 42% and would have already occurred.

B is incorrect but plausible. It is true that OS1212.01 contains guidance for performing an emergency fill from the fire protection system (OS1212.01 Attachment D), however that method is only performed when the affected PCCW train is the only one available to cool safety related equipment. The stem of the question states that all plant equipment is operating normally for 100% power, which means that Train "B" PCCW is operating. It is true that Train "A" PCCW will automatically isolate to the Containment Building if level decreases to 36%.

C is incorrect but plausible. It is true that Step 5 of OS1212.01 directs the crew to perform a local makeup to the Train "A" PCCW Head Tank from the Demineralized Water System (DM). It is also true that Train "A" PCCW will automatically isolate to the Waste Process Building if level continues to decrease, however the setpoint is however the setpoint is 42% and would have already occurred.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1212.01, "PCCW System Malfunction" |

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L1445I02 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 | 55.41 | **41.10** | |

| | | |
|---|---|---|
| | 55.43 | 43.5 |
| Comments: | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#8** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3<br>027AK2.03 - Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners | |
| | Importance Rating | 2.6 | |

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- Pressurizer pressure is at 2235 psig.
- Pressurizer pressure control is in automatic.
- Subsequently, the Master Pressurizer Pressure Controller setpoint drifts from 2235 psig to 2160 psig and system components respond.
- The Master Pressure Controller is placed in MANUAL.

What action should be taken next?

A. Open both spray valves using the spray valve controllers.

B. De-energize the backup heaters utilizing the heater control switch.

C. Depress the INCREASE pushbutton to open both spray valves and de-energize the backup heaters.

D. Depress the DECREASE pushbutton to close both spray valves and energize the backup heaters.

Proposed Answer: D

Explanation (Optional):

D is correct. When the setpoint drifts to 2160 psig this creates a pressure error signal such that measured pressure is above setpoint. This condition would result in the spray valves opening, creating a downward pressure transient below the normal system pressure of 2235psig. If the operator depresses the decrease pushbutton on the controller this will drive the controller output signal down such that the spray valves close and then the backup heaters energize. This action would then counter the downward trend in system

pressure.

A is incorrect but plausible. The student could have a misconception (backwards logic) with regard to the error signal created by the errant setpoint value. Opening the spray valves would exacerbate the situation.

B is incorrect but plausible. The student could have a misconception (backwards logic) with regard to the error signal created by the errant setpoint value. De-energizing the heaters would not correct the situation.

C is incorrect but plausible. It is correct that the operator should adjust the Master Pressure Controller output signal, but opening the spray valves would exacerbate the situation.

| | | | | |
|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • 1-NHY-509026, Pressurizer Pressure Control | | | |
| Proposed references to be provided to applicants during examination: | | | | None |
| Learning Objective: | | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | X | Seabrook TEB #26879 | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | **41.7** | | |

| | 55.43 | |
|---|---|---|

Comments:

Parent Question:

Given the following plant conditions:

1) 100% power.
2) Pressurizer pressure is 2235 psig.
3) Pressurizer pressure control is in automatic.
4) The Master Pressurizer Pressure Controller malfunctions.
5) Automatic control setpoint drifts from 2235 psig to 2160 psig, and components respond.
6) The Master Pressure Controller is placed in MANUAL.

What actions should be taken next?

A. Depress the INCREASE pushbutton. Closes both spray valves and energizes backup heaters.
B. Depress the DECREASE pushbutton. Closes both spray valves and energizes backup heaters.
C. Depress the INCREASE pushbutton. De-energizes backup heaters and opens both spray valves.
D. Depress the DECREASE pushbutton. De-energizes backup heaters and opens both spray valves.

Correct Answer: B

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#9** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000029 (EPE 29) Anticipated Transient Without Scram / 1<br>029EK2.06 - Knowledge of the interrelations between the and the following an ATWS: Breakers, relays, and disconnects. | |
| | Importance Rating | 2.9 | |

Proposed Question:

The crew performed a manual reactor trip actuation from the main control board. Reactor Trip Breaker "B" (RTB) opened however Reactor Trip Breaker "A" (RTA) did not open and cannot be opened by any means.

What is the status of the reactor and what is the effect on Safety Injection from RTA not being open?

A. The reactor is tripped. The train "A" ECCS pumps will not automatically start.

B. The reactor is not tripped. The train "A" Safety Injection signal cannot be reset.

C. The reactor is tripped. The train "A" automatic Safety Injection signal cannot be blocked.

D. The reactor is not tripped. The train "B" automatic Safety Injection signal cannot be blocked.

Proposed Answer: C

Explanation (Optional):

C is correct. The two reactor trip breakers (RTA and RTB) are configured in series with each other. If either breaker opens the reactor will trip. Each of the reactor trip breakers provides a "P-4" signal for its respective train. With regard to Safety Injection signal logic, the associated trains P-4 signal allows for a block of further automatic signals on that train. If the respective train's reactor trip breaker is not open, then the requisite P-4 signal is not generated.

A is incorrect but plausible. It is true that the reactor is tripped, as the two reactor trip breakers (RTA and RTB) are configured in series with each other. If either breaker opens the reactor will trip, however the Train "A" ECCS pumps will not be prevented from automatically starting.

B is incorrect but plausible. The reactor trip breakers (RTA and RTB) are configured in series with each other and the reactor trip bypass breakers (BYA and BYB) are configured in parallel with their respective train's reactor trip breaker. If the student has a conceptual error with regard to reactor trip breaker configuration they could incorrectly determine that the reactor is not tripped. It is true

that the reactor trip breakers provide an input to the Safety Injection signal logic, however that input (P-4) is associated with the block of additional automatic SI signals vice the ability to reset the SI signal.

D is incorrect but plausible. The reactor trip breakers (RTA and RTB) are configured in series with each other and the reactor trip bypass breakers (BYA and BYB) are configured in parallel with their respective train's reactor trip breaker. If the student has a conceptual error with regard to reactor trip breaker configuration they could incorrectly determine that the reactor is not tripped. It is true that the reactor trip breakers provide an input to the Safety Injection signal logic, however that input (P-4) is associated with the block of additional automatic SI signals vice the ability to reset the SI signal, and the P-4 signal is associated with the same train, i.e. a Train "A" P-4 signal does not provide any input to the Train "B" Safety Injection circuitry.

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|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-509042, Westinghouse Functional Diagram, Reactor Trip Signals<br>• 1-NHY-509048, , Westinghouse Functional Diagram, Safeguard Actuation Signals |

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|---|---|
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | • L8056I21<br>• L8056I29<br>• L8056I10 |

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|---|---|---|---|---|
| Question Source: | Bank # | | | |
| | Modified Bank# | X | Seabrook TEB#16307 | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 | 55.41 | **41.7** | |

| | 55.43 | |
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Comments:

Parent question:

The following plant conditions exist:

- The plant was initially at 100% power.
- A LOCA occurred resulting in dual train Safety Injection and Reactor Trip signals.
- Reactor Trip Breaker "A" (RTA) does not open, and cannot be opened by any means.
- All other systems and components function as designed.

What is the effect of RTA failing to open?

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

B. Train "A" automatic Safety Injection signals cannot be blocked.

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

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

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#10** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000038 (EPE 38) Steam Generator Tube Rupture / 3<br>038EK1.02 - Knowledge of the operational implications of the following concepts as they apply to the SGTR: Leak rate vs. pressure drop | |
| | Importance Rating | 3.2 | |

Proposed Question:

Given the following plant conditions:

- The crew is implementing E-3, "Steam Generator Tube Rupture" and is determining the required core exit thermocouple temperature prior to initiating the controlled cooldown.
- Steam Generator information is as follows:
  - "A" Steam Generator is intact and pressure is 890 psig.
  - "B" Steam Generator is ruptured and pressure is 900 psig.
  - "C" Steam Generator is intact and pressure is 1125 psig.
  - "D" Steam Generator is ruptured and pressure is 990 psig.

The required core exit thermocouple temperature target temperature for the cooldown will be based on pressure in the ____ Steam Generator.

A. A

B. B

C. C

D. D

| Proposed Answer: | B |
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Explanation (Optional):

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|---|---|---|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • E-3, "Steam Generator Tube Rupture" | | | |
| Proposed references to be provided to applicants during examination: | | | | None |
| Learning Objective: | | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.8, 41.10** | | |
| | 55.43 | | | |
| | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#11** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000055 (EPE 55) Station Blackout / 6<br>055EK3.02 - Knowledge of the reasons for the following responses as the apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power | |
| | Importance Rating | 4.3 | |

Proposed Question:

What is the basis for the strategy of depressurizing the intact Steam Generators at maximum allowable rate in procedure ECA-0.0, "Loss of All AC Power?"

A. To prevent saturation of reactor coolant by maintain subcooling greater than 40°F.
B. To minimize reactor coolant inventory loss through the reactor coolant pump seals.
C. To enhance emergency feedwater flow capability from the steam driven EFW pump.
D. To minimize core cooling challenges while forced reactor coolant flow is unavailable.

Proposed Answer: B

Explanation (Optional):

B is correct. Step 18 of ECA-0.0 prescribes depressurizing the intact Steam Generators to reduce RCS leakage. Per the Westinghouse ECA-0.0 background document the intact steam generators are depressurized "thereby reducing RCS temperature and pressure to reduce RCP seal leakage and minimize RCS inventory loss.

A is incorrect but plausible. The steam generators are depressurized in part to reduce RCS temperature, however this is pursuant to minimizing RCP seal degradation due to elevated temperature. There is no guideline in step 18 pursuant to maintaining 40°F subcooling. A note prior to step 17 states that voiding in the vessel upper head region may occur and that this should not prevent continued depressurization.

C is incorrect but plausible. Depressurizing the steam generators would reduce backpressure on the EFW lines and allow for increased EFW flow capability, however this is not included in the basis for the depressurization.

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| D is incorrect but plausible. The steam generators are depressurized in part to reduce RCS temperature however this is pursuant to minimizing RCP seal degradation due to elevated temperature. It is plausible that core cooling would be a concern when forced flow cooling is not available and the plant must rely on natural circulation. If this were true then depressurizing the steam generators would be a plausible strategy to enhance natural circulation. | | | | |
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ECA-0.0, Loss of All AC Power, Step 17<br>• Westinghouse Background Document, ECA-0.0, pgs. 7-8, 71, and 120-125. | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L8067I03, L8067I04, L8067I10 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2010 NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | 41.5, 41.10 | | |
| | 55.43 | | | |
| Comments: N/A | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#12** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000057 (APE 57) Loss of Vital AC Instrument Bus / 6<br>057AA2.04 - Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: ESF system panel alarm annunciators and channel status indicators | |
| | Importance Rating | 3.7 | |

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- Pressurizer pressure channel PT-455 failed high three (3) hours ago.
- The operating crew carried out the actions of OS1201.06, "PZR Pressure Instrument/Component Failure" and removed the channel from service.
- All required bistables have been tripped as indicated on status panels UL-1 and UL-6.
- Channel PT-457 is now the controlling channel.
- The PZR Pressure Controller has been returned to automatic control.

Subsequently, loss of 120 VAC panel PP-1C then occurs.

What is the immediate impact on the plant, if any?

A. Both Pressurizer PORVs will open.

B. Safety Injection will automatically actuate.

C. The plant will remain at 100% power with no impact on plant conditions.

D. Pressurizer control group heaters will go to minimum output and the spray valves will close.

Proposed Answer: B

Explanation (Optional):

B is correct. An automatic Safety Injection signal will occur on Pressurizer Low Pressure. The actuation signal has a 2/4 logic as sensed by pressure channels PT-455, 456, 457, and 458. The question stem states that channel PT-455 failed high and that all required bistables have been tripped. With PT-455 tripped a ¼ logic exists for the Pressurizer Pressure Low Safety Injection

signal, as would be indicated on status panel UL-1. Upon the loss of 120 VAC PP-1C the bistable for PT-457 would trip, resulting in a 2/4 logic being met, which would cause an automatic Safety Injection signal.

A is incorrect but plausible. If the student misunderstands the input and function of each of the four pressure channels into the pressure control circuity it would be conceivable that having PT-455 and PT-457 tripped would meet the interlock and actuation bistables for the PORVS. With PT-457 selected as the "controlling" channel, PT-456 would be the "backup" channel. In this case PT-457 feeds into the actuation bistable for the "A" PORV, however that PORV would have to be armed by PT-458, which remains unaffected by the conditions stated in the question stem. Additionally, if PT-457 is de-energized it would correlate to low pressurizer pressure vice high.

C is incorrect but plausible. The student may confuse the state of bistable "tripped" and bistable "bypassed". Per Tech. Specs a failed channel may be placed in bypass for up to 6 hours to allow for testing of the remaining channels, as is stated in OS1201.06. In this case, two of the remaining three channels bistables (PT-456, 457, and 458) would have to trip to cause an SI, which does not happen.

D is incorrect but plausible. It is true that the "controlling" channel provides the control signal to both spray valves and the control group heaters, however if PT-457 were de-energized by the loss of PP-1C then the pressure controller output would correlate to low pressurizer pressure, which would cause the heaters to energize and the spray valves to open.

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|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • 1-NHY-509051, Pressurizer Pressure Control Functional Diagram<br>• OS1201.06, "PZR Pressure Instrument/Component Failure" | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | • L8027I03, L8027I04, L8027I06<br>• L1182I05 | | | |
| Question Source: | Bank # | X | Seabrook TEB#20441 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | | | |
| | 55.43 | 43.5 | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#13** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000058 (APE 58) Loss of DC Power / 6<br>058AA1.03 - Ability to operate and / or monitor the following as they apply to the Loss of DC Power: Vital and battery bus components | |
| | Importance Rating | 3.1 | |

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 |
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Explanation (Optional):

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.

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| 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):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1248.01, Loss of a Vital 125 VDC Bus | | | |
| Proposed references to be provided to applicants during examination: | | | | None |
| Learning Objective: | L1189I02 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2009 NRC Remedial Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | 41.8,41.10 | | |
| | 55.43 | | | |
| Comments: | | | | |

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

Proposed Question:

Why does OS1216.01, "Degraded Ultimate Heat Sink" direct placing both ocean Service Water pumps in Pull-To-Lock and placing the associated Cooling Tower pumps in NA-Start?

A. Allows the operator to reset the Tower Actuation signal.

B. Prevents a subsequent Tower Actuation signal from reoccurring if system pressure decreases.

C. In the event of a loss of power it allows the Cooling Tower Pump discharge valve to automatically reopen when its electrical bus is re-energized.

D. In the event of a loss of power it ensures that the Cooling Tower Pump breaker remains closed and the pump starts as soon as the emergency diesel generator breaker closes.

Proposed Answer: C

Explanation (Optional):

C is correct. The Cooling Tower Pump discharge valve is interlocked with the pump breaker. The discharge valve opening circuit includes a 10 second time delay pickup relay that is energized by the pumps 52S contact.

A is incorrect but plausible. There is no Cooling Tower Pump control switch position associated with the Tower Actuation reset logic.

B is incorrect but plausible. It is true that there SW system pressure feeds into the Tower Actuation logic, however the system pressure as sensed on the discharge of the ocean SW pumps, and there is an interlock that requires that the ocean SW pumps have to have been running for greater than 28 seconds.

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| D is incorrect but plausible. It is true that the Cooling Tower Pump will restart after a loss of power, however the pump breaker does not remain closed, with the pump breaker being cycled onto the bus with the Emergency Power Sequencer HR8 stepping relay. |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-301107, Sheet CP8a<br>• OS1216.01, "Degraded Ultimate Heat Sink" |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L8037I06, L1193I02 |

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|---|---|---|---|---|
| Question Source: | Bank # | X | Seabrook TEB#11879 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |

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|---|---|---|
| 10 CFR Part 55 Content: | 55.41 | **41.8** |
| | 55.43 | |
| Comments: | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#15** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000065 (APE 65) Loss of Instrument Air / 8<br>065AA1.02 - Ability to operate and / or monitor the following as they apply to the Loss of Instrument Air: Components served by instrument air to minimize drain on system | |
| | Importance Rating | 2.6 | |

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- The crew has entered ON1242.01, "Loss of Instrument Air" based on indications of decreasing service air and instrument air pressures and is implementing procedure Step 3, "Check if Service Air Should Be Isolated".
- Service air pressure is 93 psig and lowering.

How do SA-V-92 and SA-V-93, "Service Air Isolation Valves" function and what action should the crew take?

A. SA-V-92 and 93 automatically close when service air pressure decreases to 90 psig. Per procedure Step 3, the crew should close both valves from the Main Control Board.

B. SA-V-92 automatically closes when service air pressure decreases to 95 psig. SA-V-93 automatically closes when service air pressure decreases to 90 psig. Per procedure Step 3, the crew should close both valves from the Main Control Board.

C. SA-V-92 and 93 automatically close when service air pressure decreases to 85 psig. Per procedure Step 6, "Locate and Isolate System for Air Leaks", the crew should dispatch the Secondary NSO to Control Panel 66 to CLOSE both valves.

D. SA-V-92 and 93 automatically close when service air pressure decreases to 95 psig. Per procedure Step 6, "Locate and Isolate System for Air Leaks", the crew should dispatch the Secondary NSO to Control Panel 66 to verify that both valves have closed.

| Proposed Answer: | A |
|---|---|

Explanation (Optional):

A is correct. Both valves are designed to automatically close when service air header pressure decreases to 90 psig. Per ON1242.01, "Loss of Instrument Air", step 3, "Check if Service Air Should Be Isolated" the crew is directed to close both valves if service air pressure is "Decreasing OR Less Than 90 PSIG". The valves have a common control switch which is located on the Main Control Board.

B is incorrect but plausible. It is true that both valves are designed to automatically close, however the setpoint of each valve is 90 psig. It is also true that ON1242.01, "Loss of Instrument Air", step 3, "Check if Service Air Should Be Isolated" directs the crew to close both valves if service air pressure is "Decreasing OR Less Than 90 PSIG". It is also true that the control switch for the valves is located on the Main Control Board.

C is incorrect but plausible. It is true that both valves are designed to automatically close, however the setpoint of each valve is 90 psig. It is plausible that the crew would dispatch the Secondary NSO to Control Panel 66, as that panel is located in the Turbine Building where the air system compressors and isolation valves are located, however there is no control switch for the valves on CP-66.

D is incorrect but plausible. It is true that both valves are designed to automatically close, however the setpoint of each valve is 90 psig. It is plausible that the crew would dispatch the Secondary NSO to Control Panel 66, as that panel is located in the Turbine Building where the air system compressors and isolation valves are located, however there is no position indication for the valves on CP-66.

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| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • ON1242.01, "Loss of Instrument Air"<br>• 1-NHY-503822, SA Isolation Valves V92 & V93 | | | |
| Proposed references to be provided to applicants during examination: | | | None | |
| Learning Objective: | • L8023I05<br>• L1194I02 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#16** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | 000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6<br>077AK2.06 - Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Reactor Power | |
| | Importance Rating | 3.9 | |

Proposed Question:

Given the following conditions:

- The plant is at 90% power.
- The turbine load limiter is set at 5% above current load.
- Main generator VAR load is 50 MVAR lagging.
- A disturbance on the electrical grid has caused grid voltage and frequency to decrease.

What effect will this have on reactor power and how is the reactor protected from a degradation of reactor coolant flow?

A. Reactor power will decrease. The reactor will trip if reactor coolant pump voltage drops below 10,200 volts on 1of 4 pumps.

B. Reactor power will increase. The reactor will trip if reactor coolant pump voltage drops below 10,200 volts on 2 of 2 pumps on 1 of 2 busses.

C. Reactor power will increase. The reactor will trip if reactor coolant pump frequency drops below 55.5 hz on 1of 2 pumps on 2 of 2 busses.

D. Reactor power will decrease. The reactor will trip if reactor coolant pump frequency drops below 55.5 hz on 2 of 2 pumps on 1 of 2 busses.

| | | |
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| Proposed Answer: | C | |

Explanation (Optional):

C is correct. As grid frequency decreases the plants generator amperage will increase as the machine picks up VAR loading. The reactor is protected from a degradation of flow by a Reactor Coolant Pump Low Frequency Trip; Setpoint 55.5 hz; Coincidence 1/2 pumps on 2/2 busses.

A is incorrect but plausible. The student could incorrectly determine that reactor power would decrease if they had a conceptual error regarding generator amperage loading relative to grid voltage change. There is a reactor trip based on low voltage, however its coincidence is 1/2 pumps on 2/2 busses.

B is incorrect but plausible. It is true that reactor power will increase. There is a reactor trip based on low voltage, however its coincidence is 1/2 pumps on 2/2 busses.

D is incorrect but plausible. The student could incorrectly determine that reactor power would decrease if they had a conceptual error regarding generator amperage loading relative to grid voltage change. It is true that the reactor will trip based on reactor coolant pump frequency, however the coincidence is 1/2 pumps on 2/2 busses.

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| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • E-0, "Reactor Trip or Safety Injection", Attachment L |

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| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | X | Beaver Valley 2012 | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 | 55.41 | **41.4, 41.5, 41.7, 41.10** | |

| | 55.43 | |
|---|---|---|

Comments:

Parent question:

Given the following plant conditions:

- The plant is operating at 95% power with all systems in NSA.
- Turbine control is selected to first stage IN.
- Valve position limiter is set 5% above Governor Valve Position.
- DLC Systems Operations Control Center reports that disturbances have resulted in degraded grid frequency and voltage.
- The control room team has entered AOP ½.35.1, Degraded Grid.

Given these conditions, which ONE of the following describes the relationship between degraded grid frequency/voltage and reactor power?

As grid frequency/voltage continues to drop, reactor power will _____. An automatic reactor trip will occur if 2 of 3 Reactor Coolant Pump (RCP) 4KV Bus frequencies drop to _____.

A. increase, 57.5hz

B. increase, 58.5 hz

C. be unaffected, 57.5 hz

D. be unaffected, 58.5 hz

Answer: A

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#17** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | (W E04) LOCA Outside Containment / 3<br>WE04EG2.4.4 - Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures. | |
| | Importance Rating | 4.5 | |

Proposed Question:

Given the following:

- Reactor trip with Safety Injection from 100% power.
- The crew has entered E-0, "Reactor Trip or Safety Injection."
- Containment conditions:
  - Containment Radiation Monitors indicate normal and stable.
  - Containment pressure is normal and stable.
  - Containment Building Sump B level is 3.16 inches and stable.
- RHR Vault 'A' Area Radiation Monitor RM-6538-1 is reading 1.29 E+02 mR/Hr and increasing.
- PAB -6 Foot Elevation North Area Radiation Monitor RM-6508-1 is reading 4.49 mR/Hr and stable.
- RCS conditions:
  - RCS pressure is 1750 psig and decreasing.
  - Pressurizer level is 5% and decreasing.
  - PORV's are closed.
  - Pressurizer spray valves are closed.
- Steam Generator conditions:
  - All Steam Generator pressures are approximately 950 psig and slowly decreasing.
  - Steam Generator narrow range levels all indicate off scale low.
  - Secondary radiation is normal and stable.

What procedure should be entered?

A. E-1, "Loss of Reactor or Secondary Coolant"
B. ES-1.1, "SI Termination"
C. ECA-1.2, "LOCA Outside Containment"
D. ES-1.2, "Post LOCA Cooldown and Depressurization"

| Proposed Answer: | C |
|---|---|

| Explanation (Optional): |
|---|
| C is correct. Per the conditions in the question stem there is a loss of reactor coolant that is not inside containment and not to the secondary side. The determination of a LOCA outside containment is based on radiation indications in auxiliary buildings.<br><br>A is incorrect but plausible. The conditions in the stem of the question are indicative of a loss of reactor coolant however the specific indications utilized as entry conditions for procedure E-1 are based on containment building radiation, pressure and level which are all normal.<br><br>B is incorrect but plausible. SI Termination is one of the major EOP transition points from E-0 however the ECCS termination criteria are not met as RCS pressure and level are decreasing.<br><br>D is incorrect but plausible. ES-1.2, Post LOCA Cooldown and Depressurization is one of the major procedural transitions from E-0 however the transition is based on stable pressurizer level. |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | E-0, "Reactor Trip or Safety Injection". |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L1209I04 |

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|---|---|---|---|---|
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2010 NRC Exam | | |

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

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#18** | Tier # | 1 | |
| | Group # | 1 | |
| | K/A # | (W E11) Loss of Emergency Coolant Recirculation / 4<br><br>WE11EK1.3 - Knowledge of the operational implications of the following concepts as they apply to the (Loss of Emergency Coolant Recirculation): Annunciators and conditions indicating signals, and remedial actions associated with the (Loss of Emergency Coolant Recir). | |
| | Importance Rating | 3.6 | |

Proposed Question:

Given the following conditions:

- A large break LOCA is in progress.
- There is a loss of power to Bus 6 only.
- Cold leg recirculation has been established per ES-1.3, "Transfer to Cold Leg Recirculation".
- Just after returning to E-1, "Loss of Reactor or Secondary Coolant" the RH-P-8A shaft shears.
- RWST level is 100,000 gallons.
- Containment pressure is 20 psig and slowly lowering.
- The crew enters ECA-1.1, "Loss of Emergency Coolant Recirculation".

Per ECA-1.1, what action should the crew take next?

A. Stop the "A" Charging Pump only.

B. Stop the "A" SI pump and "A" Charging Pump only.

C. Stop the "A" CBS pump, "A" SI pump, and "A" Charging Pump.

D. Stop both the Train "A" and Train "B" SI Pumps and Charging Pumps.

| Proposed Answer: | B |
|---|---|

Explanation (Optional):

B is correct. Per ECA-1.1, “Key Cautions and Notes”, if the suction source is lost to any ECCS or spray pump, the pump should be stopped. The question stem states that “Cold Leg Recirculation” has been established. ECCS system configuration for Cold Leg Recirculation has the RHR pumps aligned as the suction source for the SI pumps and Charging Pumps. In this case the suction source is lost to the Train “A” SI Pump and Charging Pump. Both of those pumps should be stopped.

A is incorrect but plausible. It is true that the train “A” Charging Pump should be stopped, as its suction source (RH-P-8A) has been lost, however the suction source to the train “”A” SI pumps has also been lost and that pump should be secured as well. There is a common operator misconception with regard to which ECCS and/or CBS pumps are supplied from their respective trains RHR pump during “Cold Leg Recirculation”.

C is incorrect but plausible. It is true that the train “A” SI and Charging Pump should be stopped, as their suction source (RH-P-8A) has been lost, however the train “A” CBS pump should not be stopped. Step 11 of ECA-1.1 does contain actions for securing CBS pumps, however that action would not occur until the crew processed the procedure to step 11.

D is incorrect but plausible. RH-V-21 does allow for a discharge cross-tie between ECCS trains, however there is no procedural action prescribed to do that, and the RHR supply line to the SI and Charging Pumps are upstream of RH-V-21.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ECA-1.1, “Loss of Emergency Coolant Recirculation” |

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |

| | |
|---|---|
| Learning Objective: | L1209I03 |

| | | | | |
|---|---|---|---|---|
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |

| | | |
|---|---|---|
| 10 CFR Part 55 Content: | 55.41 | **41.8, 41.10** |
| | 55.43 | |

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#19** | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 000001 (APE 1) Continuous Rod Withdrawal / 1<br>001AK2.06 - Knowledge of the interrelations between the Continuous Rod Withdrawal and the following: T-ave./ref. deviation meter | |
| | Importance Rating | 3.0 | |

Proposed Question:

Given the following plant conditions:

- The plant is at 75% power.
- Rod Control is AUTOMATIC
- Rods begin stepping OUT.
- The PSO places rod control in MANUAL and rod motion stops.

What condition caused rods to start stepping out?

A. Power Range channel N41 failed high.

B. RCS Loop 1 Narrow Range Tcold failed low.

C. Power Range channel N43 upper detector failed low.

D. Turbine impulse pressure instrument PT-505 failed low.

Proposed Answer: B

Explanation (Optional):

B is correct. The rod control system utilizes RCS Average Tavg as input to its "temperature error" circuit. The circuit compares Average Tavg to Tref (derived from turbine impulse pressure instrument PT-505) . With RCS Loop 1 Narrow Range Tcold failing low "Average Tavg" would indicate lower than Tref (Tavg/Tref deviation) and an outward rod motion signal would be generated.

A is incorrect but plausible. It is true that Power Range nuclear instrumentation provides an input to the "Power Mismatch" portion of rod control, however, with Power Range channel N41 failing high the power mismatch circuit would generate a signal for inward rod motion.

C is incorrect but plausible. It is true that Power Range nuclear instrumentation provides an input

| | | | | |
|---|---|---|---|---|
| to the "Power Mismatch" portion of rod control, however the circuit utilizes an "auctioneered high" NI power signal.<br><br>D is incorrect but plausible. It is true that the rod control system utilizes PT-505 as input to its "temperature error" circuit, however PT-505 failing low would create an inward rod motion signal. | | | | |
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-509049, Rod Control & Blocks Westinghouse Functional Diagram | | | |
| Proposed references to be provided to applicants during examination: | | | None | |
| Learning Objective: | L8031I05 | | | |
| Question Source: | Bank # | X | Seabrook TEB #14350 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#20** | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2<br>028AA2.05 - Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: Flow control valve isolation valve indicator | |
| | Importance Rating | 2.6 | |

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- Pressurizer level control is selected to the channel 461/460 configuration.
- An event occurs which causes the following:
  - Charging flow is 60 gpm and decreasing.
  - Pressurizer level is 63% and increasing.
  - All pressurizer heaters deenergize.
  - CS-V-145, “Letdown Regen HX Iso” indicates closed.

What event caused these conditions?

A. CS-V-145 failed CLOSED.

B. Pressurizer level channel LT-459 failed low.

C. Pressurizer level channel LT-460 failed low.

D. Pressurizer level channel LT-461 failed high.

| Proposed Answer: | C |
|---|---|

Explanation (Optional):

C is correct. LT-460 is the “backup” level channel. Failure of either the primary or backup level channel will cause CS-V-145 to close and all heaters to deenergize. With letdown flow isolated pressurizer level will increase above the 100% power setpoint of 61%. When level rises above 61% then charging flow will decrease as the controlling level channel (LT-461) inputs into the level controller.

A is incorrect but plausible. CS-V-145 is an air operated valve that fails to the CLOSED position. With CS-V-145 failed closed pressurizer level would increase and charging flow would decrease, however the pressurizer heaters would not deenergize.

B is incorrect but plausible. The student may incorrectly think that LT-459 is a non-selectable channel which causes a letdown isolation on low level. Non-selectable channel inputs do exist for pressurizer control system features; however it is associated with pressurizer pressure control system vice the level control system.

D is incorrect but plausible. If the controlling level channel fails high then charging flow will decrease, however pressurizer level would decrease. Additionally, failure high does not cause a letdown isolation or deenergization of pressurizer heaters.

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|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • 1-NHY-509027, Pressurizer Level Control, Process Control Block Diagram | | | |
| Proposed references to be provided to applicants during examination: | | | None | |
| Learning Objective: | • L8027I05<br>• L8027I06 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | | | |
| | 55.43 | **43.5** | | |

Comments:

This question addresses a common operator misconception. The students are required to understand the pressurizer level control circuitry per lesson L8027I, "PPLC". Students often have a misconception regarding the "selectability of channels that provide the letdown isolation signals. Students often have the misconception that channel L-459 is the permanent non-selectable channel that provides the isolation signal.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#21** | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 000051 (APE 51) Loss of Condenser Vacuum / 4<br>051AK3.01 - Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capability upon loss of condenser vacuum | |
| | Importance Rating | 2.8 | |

Proposed Question:

Given the following conditions:

- The plant is initially at 100% power.
- Main condenser vacuum is initially at 29" Hg and stable.
- A decrease in condenser vacuum occurs.

Assuming no operator interaction, in which order will the following actions occur?

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

A. 3,1,4,2

B. 3,4,2,1

C. 4,3,2,1

D. 4,3,1,2

Proposed Answer: B

Explanation (Optional):

B is correct. Per ON1233.01, "Loss of Condenser Vacuum", "Key Cautions and Notes", the following actions occur on decreasing vacuum:

| | |
|---|---|
| 26" | Standby Vacuum Pump starts |
| 25" | Loss of C-9, Steam Dumps blocked |
| 22.4" | Main Turbine trip |
| 18.5" | Main Feedpump trip |

Distractors A, C, and D are all plausible as they all contain actions that will occur, but not in the correct order.

| | | | | | |
|---|---|---|---|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ON1233.01, "Loss of Condenser Vacuum" | | | | |
| Proposed references to be provided to applicants during examination: | | | | None | |
| Learning Objective: | L1188I06, L8042I06 | | | | |
| Question Source: | Bank # | X | Seabrook TEB #25113 | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | | | | |
| Question History: | Last NRC Exam | n/a | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | X | |
| | Comprehension or Analysis | | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.5,41.10** | | | |
| | 55.43 | | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#22** | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 060 Accidental Gaseous Radwate Release<br>AK3.03 Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste: Actions contained in EOP for accidental gaseous-waste release. | |
| | Importance Rating | 3.8 | |

Proposed Question:

Given the following conditions:

- The BOP control room operator identifies that the following radiation monitors are in high alarm at the RDMS panel:
  - RM6503-1, "Waste Gas Compressor Inlet"
  - RM6504-1, "Waste Gas Compressor Outlet"
  - RM6502-1, Carbon Delay Bed Inlet"
- The Waste Gas System continues to operate in its normal alignment.

Per OS1252.01, "Process or Effluent High Radiation", what action should the crew take next, and why?

A. Notify Chemistry and Radiation Protection to coordinate for sampling of the Waste Gas system to confirm that the RDMS alarms are valid.

B. At the main control board, CLOSE WG-FV-1602, "Waste Gas Compressor Flow Control Valve because the valve should have automatically closed to isolate system flow.

C. Dispatch an NSO to the Waster Processing Building to secure the Waste Gas system per OS1020.01, "Waste Gas System Operation" because there is no automatic isolation signals associated with the radiation monitors.

D. Dispatch an NSO to Waste Processing Building Control Panel CP-38B to CLOSE VG-V-57, "PAB Hydrogenated Vent Header Isolation because the valve should have automatically closed to isolate the system flowpath to the plant vent.

| Proposed Answer: | B |
|---|---|

Explanation (Optional):

- Close the MSIVs and MSIV bypass valves
- Close the MSIV upstream drains
- Trip all of the reactor coolant pumps
- If evac is due to smoke or fire then place both trains of SSPS in TEST

A is incorrect but plausible. It is correct that the MSIVs should be closed and that the RCPs should be tripped, however a boric acid pump is only started as an RNO action for procedure step 6, after the control room has been evacuated.

C is incorrect but plausible. It is correct that SSPS is placed in Test, however starting a charging pump is not performed until procedure step 5, and a boric acid pump is only started as an RNO action for procedure step 6, both occurring after the control room has been evacuated.

D is incorrect but plausible. It is true that the reactor is tripped and the RCPs are tripped, however starting a charging pump is not performed until procedure step 5, after the control room has been evacuated.

<table>
<tr><td>Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number)</td><td colspan="4">• OS1200.02</td></tr>
<tr><td colspan="3">Proposed references to be provided to applicants during examination:</td><td colspan="2">None</td></tr>
<tr><td>Learning Objective:</td><td colspan="4">L8210I09</td></tr>
<tr><td>Question Source:</td><td>Bank #</td><td></td><td></td><td></td></tr>
<tr><td></td><td>Modified Bank#</td><td></td><td></td><td>(Note changes or attach Parent)</td></tr>
<tr><td></td><td>New</td><td>X</td><td></td><td></td></tr>
<tr><td>Question History:</td><td>Last NRC Exam</td><td colspan="3"></td></tr>
<tr><td rowspan="2">Question Cognitive Level:</td><td colspan="2">Memory or Fundamental Knowledge</td><td>X</td><td></td></tr>
<tr><td colspan="2">Comprehension or Analysis</td><td></td><td></td></tr>
<tr><td rowspan="2">10 CFR Part 55 Content:</td><td>55.41</td><td colspan="3">41.10</td></tr>
<tr><td>55.43</td><td colspan="3">43.5</td></tr>
<tr><td colspan="5">Comments:</td></tr>
</table>

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#24** | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | 000076 (APE 76) High Reactor Coolant Activity / 9<br>076AA1.04 - Ability to operate and / or monitor the following as they apply to the High Reactor Coolant Activity: Failed fuel-monitoring equipment | |
| | Importance Rating | 3.2 | |

Proposed Question:

With the plant at 100% power which of the following radiation monitors would go into ALARM to provide the FIRST indication that a nuclear fuel rod had developed a leak and what action should be taken due to the alarm?

A. RM-6520-1, "Letdown Radiation Monitor". Per OS1202.05, "Reactor Coolant System High Activity", Notify Chemistry to sample letdown to verify that the alarm indication is valid.

B. RM-6528, ""Wide Range Gas Monitor". Per OS1252.02, "Airborne High Radiation", Isolate Letdown flow by closing CS-V-145, "Letdown Regen Heat Exchanger Isolation Valve".

C. RM-6519-1, "Blowdown Flash Tank Outlet. Per OS1252.01, "Process or Effluent High Radiation", Verify that SB-CV-6519, "SG Blowdown Flash Tank Discharge" valve is CLOSED.

D. RM-6527A1, A2, B1, and B2, "COP (Containment Online Purge) Radiation Monitors". Per OS1252.02, "Airborne High Radiation", Verify CAP-V-1,2,3,and 4 AND COP-V-1,2,3 and 4 are all CLOSED.

Proposed Answer: A

Explanation (Optional):

A is correct. RM-6520-1, "Letdown Radiation Monitor" is designed to provide the first indication of a fuel cladding failure. Entry conditions for OS1202.05, "Reactor Coolant System High Activity" include "Abnormal RCS activity indicated by sampling or RDMS". Step 1 of the procedure directs the crew to have Chemistry sample letdown to verify that the alarm indication is valid.

B is incorrect but plausible. It is plausible that elevated activity through the Letdown portion of CVCS could eventually cause the WRGM to alarm, however the initial RDMS indication would be from the letdown rad monitor. It is true that OS1252.02, "Airborne High Radiation" does have an action to isolate potential effluent sources if the WRGM rad monitor went into alarm, however, the WRGM alarm would not be the fist indication of fuel failure and OS1252.02 would not be the applicable procedure for this condition.

C is incorrect but plausible. It is true that SB-CV-6519 would close, and should be verified closed per OS1252.01, "Process or Effluent High Radiation", however indications of high radiation from the blowdown flash tank would be indicative of a primary to secondary leak, which is the next barrier away from the fuel cladding barrier.

D is incorrect but plausible. It is true that The CAP and COP containment isolation valves would close in the event of alarm conditions on the COP rad monitors, however the COP rad monitors would not provide the first indication of fuel failure and would be indicative of an additional compromise to the RCS barrier.

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|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1202.05, "Reactor Coolant System High Activity<br>• OS1252.01, "Process or Effluent High Radiation"<br>• OS1252.02, "Airborne High Radiation" |

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|---|---|---|---|---|
| Proposed references to be provided to applicants during examination: | | | | None |
| Learning Objective: | L8024I03, L1181I09, L1187I02, L1187I07 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | | | |
| Comments: | | | | |

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|---|---|---|---|---|---|
| examination: | | | | | |
| Learning Objective: | L1203I07, L1212I09 | | | | |
| Question Source: | Bank # | | | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | X | | | |
| Question History: | Last NRC Exam | | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | | |
| | Comprehension or Analysis | X | | | |
| 10 CFR Part 55 Content: | 55.41 | | | | |
| | 55.43 | **43.5** | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#26** | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | (BW E08; W E03) LOCA Cooldown—Depressurization / 4<br>WE03EK1.1 - Knowledge of the operational implications of the following concepts as they apply to the (LOCA Cooldown and Depressurization): Components, capacity, and function of emergency systems. | |
| | Importance Rating | 3.4 | |

Proposed Question:

Given the following conditions:

- A Loss of Coolant Accident (LOCA) has occurred.
- The crew is implementing ES-1.2, "Post-LOCA Cooldown and Depressurization".
- The "A" Charging Pump has been stopped.
- All other intermediate and high head ECCS pumps are running.
- No Reactor Coolant Pumps are running.
- Pressurizer level is 40%.
- Per step 14, "Check If One SI Pump Should Be Stopped", sufficient subcooling and Pressurizer level exists, and the crew stops the "B" Safety Injection Pump.
- Immediately after stopping the pump RCS pressure begins to slowly decrease, RCS subcooling is 55°F and slowly lowering, and Pressurizer level is 39.8% and slowly lowering.

What action should the crew <u>immediately</u> take in response to the decreasing RCS pressure?

A. Manually reinitiate a Safety Injection signal.

B. Restart the "A" Charging Pump to restore RCS pressure to its previous value.

C. Restart the "B" Safety Injection Pump to restore RCS pressure to its previous value.

D. Monitor RCS subcooling and Pressurizer level to ensure that they stabilize above ECCS Reinitiation Criteria.

| Proposed Answer: | D |
|---|---|

| Explanation (Optional): |
|---|
| D is correct. When the first SI pump is stopped RCS pressure may reduce slightly and then stabilize. The ES-1.2 "Operator Action Summary" page contains the following:<br><br>ECCS Reinitiation Criteria<br>Following SI termination or ECCS flow reduction, manually align valves and start ECCS pumps as required if EITHER condition listed below occurs:<br>• RCS subcooling- LESS THAN 40°F<br>-OR-<br>• Pressurizer level-CANNOT BE MAINTAINED GREATER THAN 7%(28% FOR ADVERSE CONTAINMENT)<br><br>There is a **common operator misconception** regarding the concept of a finite RCS pressure transient when one of the ECCS pumps is secured. Students have historically misdiagnosed the transient and incorrectly determined the need to restart ECCS pumps. This knowledge item was presented to the students per LOIT program objective L1204I02, "State the basis or purpose of the following: RCS pressure requirements following ECCS flow reduction."<br><br>A is incorrect but plausible. ES-1.2 does contain ECCS reinitiation criteria, however it prescribes manually aligning valves and starting pumps as necessary, vice reinitiating a Safety Injection signal.<br><br>B is incorrect but plausible. The question stem states that one Charging Pump had been secured. The student may incorrectly determine that restarting the Charging Pump is in alignment with ECCS Reinitiation Criteria, however this would be a misapplication of the criteria associated with the **common operator misconception.**<br><br>C is incorrect but plausible. The student may incorrectly determine that restarting the Safety Injection Pump is in alignment with ECCS Reinitiation Criteria, however this would be a misapplication of the criteria associated with the **common operator misconception.** |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ES-1.2, "Post-LOCA Cooldown and Depressurization". |

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|---|---|
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L1204I02 | | | |
| Question Source: | Bank # | X | Seabrook TEB#30078 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive | Memory or Fundamental Knowledge | | | |

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|---|---|---|---|
| Level: | Comprehension or Analysis | X | |
| 10 CFR Part 55 Content: | 55.41 | **41.8, 41.10** | |
| | 55.43 | | |
| Comments: | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#27** | Tier # | 1 | |
| | Group # | 2 | |
| | K/A # | (BW E09; CE A13**; W E09 & E10) Natural Circulation/4<br><br>WE10EK2.1 - Knowledge of the interrelations between the (Natural Circulation with Steam Void in Vessel with/without RVLIS) and the following: Components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes and automatic and manual features. | |
| | Importance Rating | 3.3 | |

Proposed Question:

Given the following conditions:

- A loss of offsite power has occurred.
- The operating crew has proceeded through the appropriate EOPs and is currently implementing ES-0.2, "Natural Circulation Cooldown".
- An RCS cooldown and depressurization has been established.
- The crew is at Step 16, "Monitor for Steam Void in Reactor Vessel".

What instrumentation will the crew use to determine if voiding is present in the vessel?

A. Subcooling and RVLIS "Full Range"

B. Pressurizer Level and RVLIS "Full Range"

C. Subcooling and RVLIS "Dynamic Range"

D. Pressurizer Level and RVLIS "Dynamic Range"

Proposed Answer: B

Explanation (Optional):

B is correct per ES-0.2, Step 16

A is incorrect but plausible. It is true that Step 16 includes RVLIS "Full Range" however it does not include subcooling. Subcooling is plausible as the thermodynamic concept is tied to steam voiding.

| | 55.43 | |
|---|---|---|

Comments:

Parent Question:

Given the following conditions:

- Reactor Coolant System cooldown is in progress per ES-0.3, “Natural Circulation Cooldown with Steam Void in Vessel (With RVLIS)”.
- A steam void exists in the reactor vessel.
- Pressurizer level is 93%
- The crew is performing Step 4b, “Control PZR Level-LESS THAN 90%”

What actions are required to decrease pressurizer level to less than 90%?

A. Turn pressurizer heaters ON and repressurize the RCS.

B. Turn pressurizer heaters OFF and repressurize the RCS.

C. Turn pressurizer heaters ON and control charging and letdown as necessary.

D. Turn pressurizer heaters OFF and control charging and letdown as necessary.

Correct answer: C

C is incorrect but plausible. It is true that Step 16 includes RVLIS, however it uses full range vice dynamic range. Additionally, subcooling is plausible as the thermodynamic concept is tied to steam voiding.

D is incorrect but plausible. It is true that Step 16 includes Pressurizer Level, additionally it is true that Step 16 includes RVLIS, however it used full range vice dynamic range.

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|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • ES-0.2, "Natural Circulation Cooldown" | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L1225I14 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | X | Seabrook TEB#25180 | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 | 55.41 | **41.7** | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#28** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 003 (SF4P RCP) Reactor Coolant Pump<br>003K6.14 - Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Starting requirements | |
| | Importance Rating | 2.6 | |

Proposed Question:

Procedure OS1000.01, "Heatup From Cold Shutdown to Hot Standby", "Figure 1: Limitations and Setpoints" contains a limitation that states "The Reactor Coolant Pumps shall not be started when the reactor coolant system pressure is below 325 psig."

What is the basis for this limitation?

A. Ensures sufficient flow through the RCP #2 seal.

B. Ensures minimum differential pressure is maintained across the RCP #1 seal.

C. Prevents gas from coming out of solution and collecting in the impeller of the RCP.

D. Ensures sufficient net positive suction head (NPSH) for Reactor Coolant Pump (RCP) operation.

Proposed Answer: B

Explanation (Optional):

B is correct. Limiting the required reactor coolant system pressure to >/= 325 psig ensures the required minimum $\Delta P$ across the #1 seal of 220 psid. Below 220 psid the weight of the seal ring overrides the upward force of reactor coolant system pressure. Seal face contact would occur, resulting in seal damage.

A is incorrect but plausible. Sufficient flow to the RCP #2 seal is an operational requirement, however that requirement is ensured by maintaining appropriate Volume Control Tank pressure to ensure adequate backpressure on the RCP #1 seal.

C is incorrect but plausible. The RCP is a centrifugal pump so the possibility of gas binding/gas coming out of solution in the low pressure area of the pump impeller is a valid concept, however it is not the reason for the reactor coolant system pressure limitation.

D is incorrect but plausible. The RCP is a centrifugal pump so net positive suction head (NPSH) is a valid concept, however it is not the reason for the reactor coolant system pressure limitation.

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1000.01, "Heatup From Cold Shutdown to Hot Standby" | | | | |
| Proposed references to be provided to applicants during examination: | | | | None | |
| Learning Objective: | L1167I05, L8021I13 | | | | |
| Question Source: | Bank # | X | Seabrook TEB#10661 | | |
| . | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | | | | |
| Question History: | Last NRC Exam | | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | X | |
| | Comprehension or Analysis | | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | | |
| | 55.43 | | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#29** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 004 (SF1; SF2 CVCS) Chemical and Volume Control<br>004G2.4.6 – Knowledge of EOP mitigation strategies. | |
| | Importance Rating | 3.7 | |

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- Letdown flow is oscillating due to flashing downstream of the Regenerative Heat Exchanger.
- Regenerative Heat Exchanger letdown outlet temperature is 390°F and rising.
- The crew has entered OS1202.02, "Charging System Failure".

Which of the following could have caused this condition and what action does OS1202.02 prescribe to mitigate the condition?

A. PCCW flow to the Letdown Heat Exchanger has decreased. Close CS-V-145, "Letdown Isolation Valve".

B. CS-FCV-121, "Charging Flow Control Valve", has failed closed. Close CS-V-145, "Letdown Isolation Valve".

C. PCCW flow to the Letdown Heat Exchanger has decreased. Close CS-HCV-189 and 190, "Letdown Hand Control Valves".

D. CS-FCV-121, "Charging Flow Control Valve", has failed closed. Close CS-HCV-189 and 190. "Letdown Hand Control Valves".

Proposed Answer: B

Explanation (Optional):

B 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. If regen hx outlet temperature is greater than 380°F then OS1202.02, Step 1 RNO prescribes closing CS-V-145.

A is incorrect but plausible. It is true that OS1202.02 prescribes closing CS-V-145, and it is true that a reduction in PCCW flow to the Letdown Heat exchanger would cause an increase in letdown temperature downstream of that heat exchanger, however the flashing condition is upstream of the Letdown Heat Exchanger.

C is incorrect but plausible. It is true that a reduction in PCCW flow to the Letdown Heat exchanger would cause an increase in letdown temperature downstream of that heat exchanger, however the flashing condition is upstream of the Letdown Heat Exchanger. It is also true that closing CS-HCV-189 and 190 would stop letdown flow, however OS1202.02 prescribes closing CS-V-145.

D is incorrect but plausible. It is true that if CS-FCV-121 failed close there would be a loss of cooling and flashing would occur. It is also true that closing CS-HCV-189 and 190 would stop letdown flow, however OS1202.02 prescribes closing CS-V-145.

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| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OS1202.02, "Charging System Failure" | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L8024I03, L1445I09 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | X | Seabrook 2009 NRC Remedial Exam | (Note changes or attach Parent) |
| | New | | | |
| | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | **41.10** | | |

| | 55.43 | **43.5** |
|---|---|---|

Comments:

Parent 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.

Correct answer: D

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#30** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 004 (SF1; SF2 CVCS) Chemical and Volume Control<br>004K3.01 - Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: CRDS (automatic) | |
| | Importance Rating | 2.5 | |

Proposed Question:

Given the following plant conditions:

- The plant has been operating for a month following a refueling outage.
- The plant is at 85% power.
- The "A" Primary Component Cooling Water (PCCW) Head Tank level is 77%.
- The Volume Control Tank (VCT) level is 40%.
- Rod Control Bank D is at 210 steps.
- Rod Control is in Automatic.
- Tavg and Tref are matched.
- All other systems and controls are in normal alignment.
- A tube leak develops in the "A" Seal Water Return Heat Exchanger.

What will the plant response be if no operator action is taken?

A. VCT level will rise to 100%.

B. The "A" PCCW Head Tank level will slowly rise to 100%.

C. Control rods will step in to maintain Tavg and Tref approximately equal.

D. Control rods will step out to the full out position in an attempt to maintain Tavg and Tref approximately equal.

Proposed Answer: C

Explanation (Optional):

C is correct. PCCW system pressure is at approximately 100 psig. Pressure in the seal return lines is dependent on VCT pressure, which normally is approximately 10-12 psig. A leak in the "A" Seal Water Return Heat Exchanger would result in unborated water from

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| the PCCW system entering the charging portion of the CVCS system downstream of VCT. The unborated water would ultimately be delivered to the Reactor Coolant System, causing a dilution of boron concentration. The dilution would eventually result in an increasing Tavg. With Control Rods in automatic rods would begin to insert when Tavg/Tref reached +1.5°F.<br><br>A is incorrect but plausible. It is true that the VCT level would rise due to the influx of PCCW fluid into the charging pump suction line, however the VCT controls would cause VCT to divert prior to level ever reaching 100%.<br><br>B is incorrect but plausible. If the CVCS charging system pressure were greater than PCCW system pressure then the leak would be into the PCCW system, with head tank level rising. The would be the case if the seal return line were downstream of the charging pump.<br><br>D is incorrect but plausible. It is true that a leak in the “A” Seal Water Return Heat Exchanger would result in water from the PCCW system entering the charging portion of the CVCS system downstream of VCT, however the PCCW fluid is unborated, which results in a dilution event. Additionally, rods could not step out to the full out position as auto rod withdrawal is blocked at 223 steps (C-11). |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • PID-1-CS-B20720, Chemical and Volume Control System Overview<br>• PID-1-CC-B20206, Primary Component Cooling Loop A Detail |

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| Proposed references to be provided to applicants during examination: | None |

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| Learning Objective: | L8024I02, L8024I08, L8031I05 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | X | Ginna 2008 | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |

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| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 | 55.41 | **41.7** | |

| | 55.43 | |
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Comments:

Parent question:

Plant conditions are as follows:

The plant has been operating for two weeks following a refueling outage.

The reactor is currently at 100%.

CCW Surge Tank is 50%.

VCT level is 25%.

Rod Control is in automatic.

Tref and Tavg are matched.

Tube leaks develop in the Seal Water Return Heat Exchanger

What will the plant response be if no operator action is taken?

A. VCT level will slowly rise to 100%.
B. CCW Surge Tank will slowly rise to 100%.
C. Rods will step in to maintain Tavg and Tref approximately equal.
D. Rods will step out to maintain Tavg and Tref approximately equal.

Correct answer: C

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#31** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 005 (SF4P RHR) Residual Heat Removal<br>005K5.05 - Knowledge of the operational implications of the following concepts as they apply the RHRS: Plant response during "solid plant": pressure change due to the relative incompressibility of water | |
| | Importance Rating | 2.7 | |

Proposed Question:

Given the following conditions:

- The Reactor Coolant System has been placed in a water solid condition per OS 1000.04, "Plant Cooldown from Hot Standby to Cold Shutdown", Section 4, "Place the Reactor Coolant System in a Solid Condition with a Reactor Coolant Pump in Service".
- The Train "A" Residual Heat Removal System is providing RHR Letdown.
- CS-PK-131"Letdown Back Pressure Control Valve" controller is in Manual at 10% output.

Which one of the following will cause Reactor Coolant System pressure to decrease?

A. Adjusting the output of CS-PK-131, CS-HCV-128, "RHR Flow Control Valve" controller to 8%

B. A loss of instrument air to CS-HCV-128, "RHR Flow Control Valve"

C. A loss of instrument air to CS-FCV-121, "Charging Flow Control Valve"

D. Adjusting the output of CS-PK-131, CS-HCV-128, "RHR Flow Control Valve" controller to 12%

Proposed Answer: D

Explanation (Optional):

D is correct. During water solid conditions Reactor Coolant System pressure is a function of letdown flow versus charging flow. The question stem states that RHR Letdown is in service. The configuration of RHR Letdown is such that CS-HCV-128, "RHR Flow Control Valve" is upstream of CS-FCV-131, "Letdown Back Pressure Control Valve". Raising the output signal of its controller would reduce the backpressure imposed on RHR Letdown

flow, resulting in a decrease in RCS pressure.

A is incorrect but plausible. It is true that adjusting the output of CS-FCV-131 will effect pressure, however lowering the output signal to CS-FCV-131 would increase the backpressure imposed on RHR Letdown flow, resulting in an increase in RCS pressure.

B is incorrect but plausible. CS-HCV-128 fails closed on a loss of air. The student would choose this answer if they incorrectly believed that the valve failed open, in which case they would incorrectly determine that RHR Letdown flow increased, resulting in a decrease in RCS pressure.

C is incorrect but plausible. CS-FCV-121 fails open on a loss of air. The student would choose this answer if they incorrectly believed that the valve failed closed, in which case they would incorrectly determine that Charging flow decreased, resulting in a decrease in RCS pressure.

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| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OS1000.04, ""Plant Cooldown From Hot Standby to Cold Shutdown" | | | | |
| Proposed references to be provided to applicants during examination: | | | None | | |
| Learning Objective: | L3010I11 | | | | |
| Question Source: | Bank # | | | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | X | | | |
| Question History: | Last NRC Exam | | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | | |
| | Comprehension or Analysis | X | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.5** | | | |
| | 55.43 | | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#32** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 006 (SF2; SF3 ECCS) Emergency Core Cooling<br>006K1.14 - Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: IAS | |
| | Importance Rating | 3.0 | |

Proposed Question:

Which one of the following correctly describes the impact that the loss of instrument air would have on Safety Injection/ECCS alignment?

A. RH-V-14, "RHR Train A to Cold Legs 1&2" would fail to the OPEN position.

B. SI-V-112, "SI Train A Discharge Cross Connect" would fail to the CLOSED position.

C. CS-V-143, "Charging to Regen HX Isolation" would fail to the OPEN position.

D. CS-V-150, "Letdown Heat Exchanger ORC Isolation" would fail to the CLOSED position.

| Proposed Answer: | D |
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Explanation (Optional):

D is correct. CS-V-150 is an air operated valve and fails to the CLOSED position on a loss of air. Additionally, the valve receives a CLOSE signal on SI.

A is incorrect but plausible. It is true that RH-V-14 should be in the open position however the valve is a motor operated valve an unaffected by the loss of instrument air.

B is incorrect but plausible. It is true that SI-V-112 should be in the open position however the valve is a motor operated valve an unaffected by the loss of instrument air.

C is incorrect but plausible. It is true that CS-V-143 is required to close on a Safety Injection signal however the valve is a motor operated valve an unaffected by the loss of instrument air.

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| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • ON1242.01, "Loss of Instrument Air", Attachment B: Air Operated Valve Fail Positions on Loss of Instrument Air | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L8034I10 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.2 to 41.9** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#33** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 007 Pressurizer Relief Tank/Quench Tank System (PRTS)<br>A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the P S; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Overpressurization of the Pressurizer | |
| | Importance Rating | 3.6 | |

Proposed Question:

Given the following conditions:

- The plant was originally at 100%power.
- The crew manually tripped the reactor due to loss of the "A" Main Feedwater Pump.
- The "A" Pressurizer PORV opened and has failed to close.
- Safety Injection automatically actuated.
- The crew has entered E-0, Reactor Trip or Safety Injection and is performing Step 9, "Check RCS Isolated.
- The control board operator took the "A" PORV control switch to close but the PORV did not close.
- Subsequently, the control board operator took the "A" PORV Block Valve control switch to CLOSE.
- The following Pressurizer Relief Tank indications exist:
  - Level is RISING.
  - Pressure is 0 psig.

What is the expected PORV tailpipe temperature and what action should the crew take next?

A. 210°F; The PORV Block Valve is CLOSED. Continue to process E-0, "Reactor Trip or Safety Injection.

B. 210°F; The PORV Block Valve is NOT CLOSED. Transition to E-1, "Loss of Reactor or Secondary Coolant".

C. 332°F; The PORV Block Valve is CLOSED. Continue to process E-0, "Reactor Trip or Safety Injection.

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| D. 332°F; The PORV Block Valve is NOT CLOSED. Transition to E-1, "Loss of Reactor or Secondary Coolant". | | |
| Proposed Answer: | B | |
| Explanation (Optional): | | |

B is correct. PRT pressure at 0 psig indicates that the rupture disk has blown and tank pressure is low. 210°F is the equivalent saturation temperature for the given PRT temperature. PORV tailpipe temperature will correlate to PRT pressure. The given conditions in the question stem indicate that the PORV Block Valve did not close. E-0, Step 9, directs a transition to E-1 if the PORV Block Valve does not close.

A is incorrect but plausible. 210°F is the correct tailpipe temperature, however the leak has not been isolated.

C is incorrect but plausible. The PRT rupture disk is designed to rupture at 91 psig. 332°F is a plausible temperature as it is the saturation temperature for 91 psig, however the information in the stem of the question indicates that the disk has ruptured. Additionally, the leak has not been isolated.

D is incorrect but plausible. It is true that the block valve has not closed. The PRT rupture disk is designed to rupture at 91 psig. 332°F is a plausible temperature as it is the saturation temperature for 91 psig, however the information in the stem of the question indicates that the disk has ruptured.

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| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • E-0, "Reactor Trip or Safety Injection |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L8022I10, L1202I02, L1202I08 |

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| Question Source: | Bank # | X | Point Beach 2012 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 | 55.41 | **41.5** | |

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| | 55.43 | **43.5** |
| Comments: | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#34** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 008 (SF8 CCW) Component Cooling Water<br>008A2.07 - Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of high or low CCW flow rate and temperature; the flow rate at which the CCW standby pump will start | |
| | Importance Rating | 2.5 | |

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- The Primary Component Cooling Water System is in normal alignment.
- The primary control board operator (PSO) identifies the following conditions:
  - Hardwire annunciator/alarm MM-UA051, E-6, “PCCW Train B Flow Lo” is in ALARM.
  - CC-P-11B, “PCCW Pump B” has tripped.
  - 40 seconds have gone by since the alarm and pump trip occurred.

Which of the following describes the expected PCCW system response and required crew action?

A. CC-P-11D, “PCCW Pump D” should have automatically started 30 seconds after CC-P-11B tripped. Per OS1212.01, “PCCW System Malfunction”, Step 2, the crew should manually start CC-P-11D.

B. CC-P-11D, “PCCW Pump D” should automatically start 60 seconds after CC-P-11B tripped. Per OS1212.01, “PCCW System Malfunction”, Step 2, the crew should manually start CC-P-11D if it does not automatically start.

C. CC-P-11D, “PCCW Pump D” should have automatically started when the “PCCW Train B Flo Lo” alarm occurred. Per the hardwire alarm response procedure for the “PCCW Train B Flow Lo” alarm, the crew should manually start CC-P-11D.

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| D. CC-P-11D, "PCCW Pump D" is locked out. Per the hardwire alarm response procedure the crew should dispatch a NSO to locally open the breaker for Circuit #12 on the MCC-621 distribution panel. Once the breaker is open then the crew should manually start CC-P-11D. |

| Proposed Answer: | A | |
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Explanation (Optional):

A is correct. The automatic start feature for the standby pump is based on breaker position from the running pump. There is a 30 second timed delay associated with the auto start feature. A "PCCW System Low Flow" alarm is one of the entry conditions for OS1212.01, "PCCW System Malfunction". Step 2 of the procedure directs the crew to verify that there is one pump running in each PCCW loop, and if not then manually start the standby pump.

B is incorrect but plausible. It is true that the standby pump will automatically start based on breaker position from the running pump. However the timed delay associated with the auto start feature is 30 seconds vice 60. It is also true that OS1212.01 directs manually starting the standby pump.

C is incorrect but plausible. It is true that the standby pump has an auto start feature however it is based on breaker position vice from a low system flow signal. It is true that the crew should manually start the standby pump, however the hardwire alarm procedure directs the crew to implement OS1212.01.

D is incorrect but plausible. It is true that the crew should manually start the standby pump however there is no lockout feature based on system flow. There is a lockout feature based on high system temperature, and the distractor lists the correct circuit for the temperature lockout.

| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OS1212.01, PCCW System Malfunction<br>• 1-NHY-310895, sheet A58b, PCCW Pump Close Schematic |
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| Proposed references to be provided to applicants during examination: | None |
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| Learning Objective: | L8036I07, L1445I02 |
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| Question Source: | Bank # | | | |
|---|---|---|---|---|
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |

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| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | **41.5** | |
| | 55.43 | **43.5** | |
| Comments: | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#35** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 010 (SF3 PZR PCS) Pressurizer Pressure Control<br>010K1.07 - Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: Containment | |
| | Importance Rating | 2.9 | |

Proposed Question:

Given the following conditions:

- The plant is operating at 100% power.
- The Pressurizer pressure control system is operating in automatic with pressure instruments 455/456 selected for control/backup.

Assuming no operator action, which one of the following Pressurizer pressure instrument failures would result in RM6526, "Containment Particulate Radiation Monitor" going into alarm?

A. **ONLY** pressure instrument 456 failing low

B. **ONLY** pressure instrument 455 failing low

C. Pressure channel 455 failing low **OR** pressure channel 456 failing low

D. Pressure channel 455 failing low **OR** pressure channel 457 failing high

Proposed Answer: B

Explanation (Optional):

B is correct. Instrument 455 is selected as the controlling channel. The controlling channel provides input into the pressurizer spray valves, control group heaters, and PCV-456A, "PORV A". If instrument 455 fails low the control group heaters would energize and the spray valves and "A" PORV would never open. The "B" PORV receives its open signal from the backup instrument (in this case instrument 456) provided it also receives an "open interlock" signal from instrument 457 (hardwired). Given the conditions in the question stem, the "B" PORV would open. Discharge from the "B" PORV is routed to the Pressurizer Relief Tank. The PRT is protected from overpressure by rupture disc which relieve to the containment environment. When the rupture disks eventually relieve pressure then RM6526, "Containment Particulate Radiation Monitor" would go into alarm.

A is incorrect but plausible. If the student understood the effects of a failed instrument on

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| heaters and sprays but had a misconception of the instrument requirements for opening the PORVs then may choose this distractor.<br><br>C is incorrect but plausible. If the student understood the effects of a failed instrument on heaters and sprays but had a misconception of the instrument requirements for opening the PORVs then may choose this distractor.<br><br>D is incorrect but plausible. If the student understood the effects of a failed instrument on heaters and sprays but had a misconception of the instrument requirements for opening the PORVs then may choose this distractor. |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1201.06, "Pressurizer Pressure Instrument/Component Failure"<br>• 1-NHY-509026, Pressurizer Pressure Control |
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L8027I14 | | | |
| Question Source: | Bank # | X | McGuire 2008 NRC Exam | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | McGuire 2008 | | |

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| Question Cognitive Level: | Memory or Fundamental Knowledge | |
| | Comprehension or Analysis | X |

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|---|---|---|
| 10 CFR Part 55 Content: | 55.41 | **41.2 to 41.9** |
| | 55.43 | |

Comments:

**This question is associated with a common operator misconception associated with the controlling, actuating, and interlock functions of the four pressurizer pressure channels. This misconception is illustrated by the fact that there is a note in OS1201.06, "Pressurizer Pressure Instrument/Component Failure" that provides associated technical clarification with regard to selecting control/backup channels.**

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#36** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 012 (SF7 RPS) Reactor Protection<br>012A3.05 - Ability to monitor automatic operation of the RPS, including: Single and multiple channel trip indicators | |
| | Importance Rating | 3.6 | |

Proposed Question:

Given the following plant conditions:

- The plant is initially at 100% power.
- An inadvertent Safety Injection occurs.
- The Train 'B' Reactor Trip Breaker will not open.
- All other systems and equipment respond as expected.
- Per E-0, "Reactor Trip or Safety Injection" the crew has reset the Safety Injection signal on both trains.

What is the status of the Safety Injection circuitry?

A. Both trains of SI are reset. SI automatic actuation is blocked on both trains.

B. Only Train 'A' SI is reset. Only Train 'A' SI automatic actuation is blocked.

C. Both trains of SI are reset. Only Train 'A' SI automatic actuation is blocked.

D. Neither trains SI signal is reset. SI automatic actuation is not blocked on either train.

Proposed Answer: C

Explanation (Optional):

C is correct. A train specific P-4 signal allows for blocking of further automatic SI actuations once that train's SI signal is reset. Failure of Reactor Trip Breaker 'B' to open will inhibit the ability to block a further automatic SI on Train 'B' once SI is reset. The SI reset is still functional since the reset signal comes from dual train switches that do not have a P-4 interface.

A is incorrect but plausible. Both trains of SI will reset as the SI switches are dual train and have no P-4 interface. It is plausible that the auto SI signals would be blocked on both trains if the blocking signal were dual train however that signal is train specific with a P-4 signal being generated from the specific train related reactor trip breaker actuation.

B is incorrect but plausible. It is plausible that both the SI reset and auto SI block would both require a P-4 signal input however both trains of SI will reset as the SI switches are

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| dual train and have no P-4 interface.<br><br>D is incorrect but plausible. It is plausible that the P-4 signal would be dual train with both train related P-4 signals required to be reset. | | | | | |
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-509042, Reactor Trip Signals w/ Functional Diagrams<br>• 1-NHY-509048, Safeguard Actuation Signals w/ Functional Diagrams | | | | |
| Proposed references to be provided to applicants during examination: | | | | None | |
| Learning Objective: | L8056I29 | | | | |
| Question Source: | Bank # | X | Seabrook 2003 Company Exam | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | | | | |
| Question History: | Last NRC Exam | | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | | |
| | Comprehension or Analysis | X | | | |
| 10 CFR Part 55 Content: | 55.41 | 41.5 | | | |
| | 55.43 | 43.5 | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#37** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 013 (SF2 ESFAS) Engineered Safety Features Actuation<br>013K2.01 - Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control | |
| | Importance Rating | 3.6 | |

| Proposed Question: | |
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Given the following conditions:

- The plant is initially at 100% power.
- A loss of 120 Vital AC Power Panel 1A occurs
- Prior to reenergizing Power Panel 1A a Safety Injection Signal and Loss of Offsite Power event occurs.
- Both Emergency Diesel Generators repower their respective electrical busses and both Emergency Power Sequencers function properly.

Which of the following correctly lists all of the Emergency Core Cooling System (ECCS) pumps that will be running?

A. 1 Charging Pump, 1 SI Pump, 1 RHR Pump

B. 2 Charging Pumps, 1 SI Pump, 1 RHR Pump

C. 2 Charging Pumps, 2 SI Pumps, 2 RHR Pumps

D. No Charging Pump, 2 SI Pumps, 2 RHR Pumps

| Proposed Answer: | B |
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Explanation (Optional):

B is correct. Both Charging Pumps will start via the Emergency Power Sequencer and associated Loss of Power contacts on their breaker starting circuit. The Train "B" SI and RHR Pumps will start when sequenced by the Emergency Power Sequencer. The Train "A" SI and RHR Pumps will not start because PP-1A is the power source to the Train "A" ESFAS slave relays, which must be energized to actuate their associated equipment.

A is incorrect but plausible. It is true that there will be 1 SI Pump and 1 RHR pump,

however both Charging Pumps will start via the Emergency Power Sequencer and associated Loss of Power contacts on their breaker starting circuit. The student may incorrectly believe that the Train "A Charging Pump needed its associated ESFAS slave relay energized.

C is incorrect but plausible. It is true that both Charging Pumps will start via the Emergency Power Sequencer and associated Loss of Power contacts on their breaker starting circuit. The student may incorrectly believe that the ESFAS slave relays have dual power supplies, which is not the case.

D is incorrect but plausible. The student may incorrectly believe that the Charging Pumps do not restart on a Loss of Power. The student may incorrectly believe that the SI and RHR Pumps will start via the Emergency Power Sequencer and associated Loss of Power contacts on their breaker starting circuit.

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • Print 310887, Sheet A57b, RH-P-8A Close Schematic<br>• Print 310890, Sheet A56b, SI-P-6A Close Schematic<br>• Print 310891, Sheet A62b, CS-P-2A Close Schematic |
| Proposed references to be provided to applicants during examination: | None |

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| Learning Objective: | L8056I07 | | | |
| Question Source: | Bank # | X | Seabrook TEB #22064 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#38** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 022 (SF5 CCS) Containment Cooling<br>022A3.01 - Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation | |
| | Importance Rating | 4.1 | |

Proposed Question:

Given the following plant conditions:

- The plant was initially at 100% power.
- A LOCA is in progress.
- Reactor Trip and Safety Injection have occurred.
- All safeguards systems are functioning as designed.
- Containment Pressure is 24 psig and trending down slowly.

Which of the following describes the status of the Control Rod Drive Mechanism Cooling Fans and the Containment Recirculation Fans?

A. Control Rod Drive Mechanism Cooling Fans are RUNNING, Containment Recirculation Fans are operating in the FILTER MODE.

B. Control Rod Drive Mechanism Cooling Fans are TRIPPED, Containment Recirculation Fans are operating in the FILTER MODE.

C. Control Rod Drive Mechanism Cooling Fans are RUNNING, Containment Recirculation Fans are operating in the RECIRC MODE.

D. Control Rod Drive Mechanism Cooling Fans are TRIPPED, Containment Recirculation Fans are operating in the RECIRC MODE.

Proposed Answer: D

Explanation (Optional):

D is correct. Given the conditions listed a ‘P’ signal (Hi-3, 18 psig containment pressure) will have occurred. A ‘P’ signal will trip the Control Rod Drive Mechanism Fans, and will cause the Containment Recirculation Fan subsystem to run in the recirculation mode.

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| A is incorrect but plausible. It is plausible that all containment subsystem cooling would remain in service during an accident condition to aid in maintaining containment integrity however this is not part of the subsystem design. Additionally, it is plausible that the containment structure recirculation fans would run in the filter mode on a 'P' signal so as to filter post-accident products, however the subsystem actually runs in the recirc mode in order to recirculate the containment atmosphere to control hydrogen concentration.<br><br>B is incorrect but plausible. It is plausible that the containment structure recirculation fans would run in the filter mode on a 'P' signal so as to filter post-accident products, however the subsystem actually runs in the recirc mode in order to recirculate the containment atmosphere to control hydrogen concentration.<br><br>C is incorrect but plausible. A 'P' signal will cause the Containment Recirculation Fan subsystem to run in the recirculation mode. Additionally, it is plausible that all containment subsystem cooling would remain in service during an accident condition to aid in maintaining containment integrity however this is not part of the subsystem design. |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-503204, CAH-Containment Structure Recirc Filter Fan Logic Diagram<br>• 1-NHY-503202, CAH-CRDM Cooling Unit Fans Logic Diagram |

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| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L8038I04 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | X | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2010 NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | 41.7 | | |

| | 55.43 | |
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Comments:

NOTE: The question has been modified from the parent question so as to avoid technical overlap with **Question #39.**

Parent Question:

Given the following plant conditions:

- The plant was initially at 100% power.
- A LOCA is in progress.
- Reactor Trip and Safety Injection have occurred.
- All safeguards systems are functioning as designed.
- Containment Pressure is 24 psig and trending down slowly.

Which of the following describes the status of Containment Cooling Systems?

A. Control Rod Drive Mechanism Cooling Fans are RUNNING, Containment Structure Cooling Fans are RUNNING, Containment Recirculation Fans are operating in the FILTER MODE.

B. Control Rod Drive Mechanism Cooling Fans are TRIPPED, Containment Structure Cooling Fans are TRIPPED, Containment Recirculation Fans are operating in the FILTER MODE.

C. Control Rod Drive Mechanism Cooling Fans are RUNNING, Containment Structure Cooling Fans are RUNNING, Containment Recirculation Fans are operating in the RECIRC MODE.

D. Control Rod Drive Mechanism Cooling Fans are TRIPPED, Containment Structure Cooling Fans are TRIPPED, Containment Recirculation Fans are operating in the RECIRC MODE.

Correct Answer: D

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#39** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 022 (SF5 CCS) Containment Cooling<br>022K2.01 - Knowledge of power supplies to the following: 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

Explanation (Optional):

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.

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|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • 1-NHY-310931 Sheet AC5b/L803b | | | |
| Proposed references to be provided to applicants during examination: | | | None | |
| Learning Objective: | L8020I08, L8020I09, L8038I04 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2009 NRC Remedial Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#40** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 026 (SF5 CSS) Containment Spray<br>026G2.1.7 - 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 Large Break LOCA has occurred.
- The crew has entered E-1, 'Loss of Reactor or Secondary Coolant'.
- Safety Injection has been RESET.
- RWST level is 118,000 gallons and decreasing.

What is the status of the ECCS suction source?

A. RWST level has reached the semi-automatic swapover setpoint. CBS-V-8 and CBS-V-14, Containment Recirc Sump Suction 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 and CBS-V-8 and CBS-V-14, Containment Recirc Sump Suction Valves, automatically open.

C. RWST level has not reached the semi-automatic swapover setpoint. CBS-V-8 and CBS-V-14, Containment Recirc Sump Suction Valves must be manually opened per ES-1.3, 'Cold Leg Recirculation'.

D. RWST level has reached the semi-automatic swapover setpoint. CBS-V-8 and CBS-V-14, Containment Recirc Sump Suction Valves, should have automatically opened because the SI signal is still present to the CBS suction swapover circuit.

| Proposed Answer: | D |
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Explanation (Optional):

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

with SI input to swapover circuit.

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. It is true that the valves will automatically open, however the RWST swapover setpoint is at 120,478 gallons so they should be open.

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

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-503258, CBS-ECCS/Spray Recirc Signal Generation<br>• 1-NHY-503252, CBS-Cont. Sump Valves Logic Diagram |

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| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L8035I03, L8035I13 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2009 Remedial Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | | | |

Comments: Question was editorially modified so as to better serve the intent of the assigned K/A. The modification is not significant enough to meet the criteria for “significantly modified”.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#41** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 039 (SF4S MSS) Main and Reheat Steam<br>039A4.04 - Ability to manually operate and/or monitor in the control room: Emergency feedwater pump turbines | |
| | Importance Rating | 3.8 | |

| Proposed Question: | |
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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 | |
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Explanation (Optional):

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.

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

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| Driven EFW Pump would not be available as it is powered from Bus 6, and the question stem states that SEPS is not available. | | | | | |
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-310844, Sheet A80, 1-P-37B<br>• 1-NHY-310844, Sheet A93, 1-P-113 | | | | |
| Proposed references to be provided to applicants during examination: | | | | None | |
| Learning Objective: | L8045I01 | | | | |
| Question Source: | Bank # | X | | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | | | | |
| Question History: | Last NRC Exam | Seabrook 2009 Remedial Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | | |
| | Comprehension or Analysis | X | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | | |
| | 55.43 | | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#42** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 059 (SF4S MFW) Main Feedwater<br>059A2.05 - Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture in MFW suction or discharge line | |
| | Importance Rating | 3.1 | |

Proposed Question:

Given the following:

- The plant is at 100% power.
- All plant systems are in normal alignment.
- A rupture occurs in the suction line to the “A” Main Feedwater pump resulting in the following conditions:
    - “A” Main Feedwater Pump suction pressure dropped to 240 psig for 10 seconds and is now at 260 psig and slowly increasing.
    - “B” Main Feedwater Pump suction pressure dropped to 240 psig for 20 seconds and is now at 255 psig and slowly increasing.

Which of the following describes the expected impact and what action should be taken?

A. The standby Condensate Pump has automatically started. Both Main Feedwater Pumps have tripped. The crew should manually trip the reactor and enter E-0, “Reactor Trip or Safety Injection”.

B. The standby Condensate Pump has automatically started. The “A” Main Feedwater Pump has tripped. The crew should implement OS1231.03, “Turbine Runback/Setback”. Per Step 6 if Steam Generator level can NOT be maintained greater than 25% the crew should trip the reactor.

C. The standby Condensate Pump has automatically started. Neither Main Feedwater Pump has tripped. The crew should implement OS1290.02, “Response to Secondary System Transients”. Per Step 5 “Check Standby Condensate NOT Running” the crew should adjust CO-FK-4042, “Condensate Minimum Flow Controller” setpoint as

required for 3 pump operation.

D. The standby Condensate Pump has NOT started. Neither Main Feedwater Pump has tripped. The crew should implement OS1290.02, “Response to Secondary System Transients”. Per Step 4 “Monitor Main Feed Pump Suction Pressure”, the crew should start the standby Condensate Pump if Feed Pump suction pressure drops to less than 220 psig.

| Proposed Answer: | C |
|---|---|

Explanation (Optional):

C is correct. The standby Condensate Pump will automatically start if either Main Feedwater Pump suction pressure drops below 250 psig (no time delay associated with the setpoint). Neither Main Feedwater Pump would have tripped as the trip setpoints are:

- Pump “A”: <220 psig for > 6 seconds
- Pump “B”: <220 psig for > 12 seconds

Additionally per OS1290.02, Step 5, the crew should adjust CO-FK-4042, “Condensate Minimum Flow Controller” setpoint as required for 3 pump operation.

A is incorrect but plausible. It is true that the standby Condensate Pump will automatically start if either Main Feedwater Pump suction pressure drops below 250 psig (no time delay associated with the setpoint), however neither Feedwater Pump will have tripped.

B is incorrect but plausible. It is true that the standby Condensate Pump will automatically start if either Main Feedwater Pump suction pressure drops below 250 psig (no time delay associated with the setpoint), however the “A” Main Feedwater Pump would not have tripped.

D is incorrect but plausible. It is true that neither Main Feedwater Pump will have tripped, however the standby Condensate Pump should have started as the automatic start setpoint is </= 250 psig (no time delay associated with the setpoint).

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ON1035.10, “Main Feed Pump Standby and Startup Operation”, Figure 1, “Limitations and Setpoints, Item 10.<br>• OS1231.03, , “Turbine Runback/Setback”<br>• OS1290.02, “Response to Secondary System Transients” |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L8062I08, L1191I08 |

| Question Source: | Bank # | | | |
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| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |

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| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |

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| 10 CFR Part 55 Content: | 55.41 | **41.5** |
| | 55.43 | **43.5** |
| Comments: | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#43** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 059 (SF4S MFW) Main Feedwater<br>059A3.06 - Ability to monitor automatic operation of the MFW, including: Feedwater isolation | |
| | Importance Rating | 3.2 | |

Proposed Question:

The plant was at 100% power when the following events occur:

- The "C" Main Feedwater Regulating Valve fails open.
- The crew manually trips the reactor.
- Reactor Trip Breaker "B" fails to open.
- "C" Steam Generator narrow range level instruments are indicating:
    - 92% and increasing
    - 89% and increasing
    - 91% and increasing
    - 88% and increasing
- Reactor Coolant System Tavg instruments are indicating:
    - 559°F and decreasing
    - 559°F and decreasing
    - 553°F and decreasing
    - 558°F and decreasing

What is the status of the Feedwater Isolation Signal (FWI) and its basis?

A. Neither train FWI has actuated because no setpoints have been reached.

B. Both trains of FWI have actuated based on "P-14".

C. Only the Train "A" FWI signal has actuated based on "P-4 combined with P-14".

D. Only the Train "A" FWI signal has actuated based on "P-4 combined with a low Tavg".

| Proposed Answer: | B | |
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| Explanation (Optional): |
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| B is correct. The setpoint for P-14 is 2 of 4 Steam Generator narrow range levels >90.8% on any Steam Generator. There is no P-4 input associated with the P-14 logic.<br><br>A is incorrect but plausible. It is true that FWI would not have actuated based on "P-4 combined with Lo Tavg" as the setpoint is < 557°F (2 of 4 logic). A FWI signal is generated from any Safety Injection signal, so it is true that there was no SI signal. The signal would, however, be generated based on P-14. The student could choose answer "A" if they had a misconception regarding the P-14 setpoint. **NOTE: There is a common operator misconception regarding FWI signals, particularly the P-14 and Low Tavg setpoints and the association with P-4.**<br><br>C is incorrect but plausible It is true that the P-14 setpoint has been reached, however the P-14 signal logic does not include input from P-4. **NOTE: There is a common operator misconception regarding FWI signals, particularly the P-14 and Low Tavg setpoints and the association with P-4.**<br><br>D is incorrect but plausible. It is true that based on "P-4 combined with a low Tavg" that only the Train "A" signal would actuate as the "B" Reactor Trip Breaker failed to open and generate a P-4 signal for that train, however the "P-4 combined with a low Tavg" setpoint was not reached. **NOTE: There is a common operator misconception regarding FWI signals, particularly the P-14 and Low Tavg setpoints and the association with P-4.** |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-509053, FW Control and Isolation Functional Diagram" |

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| Proposed references to be provided to applicants during examination: | None |

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| Learning Objective: | | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | **41.7** | | |

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|---|---|---|
| | 55.43 | |
| Comments: | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#44** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 061 (SF4S AFW) Auxiliary/Emergency Feedwater<br>061K5.01 - Knowledge of the operational implications of the following concepts as they apply to the AFW: Relationship between AFW flow and RCS heat transfer | |
| | Importance Rating | 3.6 | |

Proposed Question:

A plant startup is in progress per OS1000.02, "Plant Startup from Hot Standby to Minimum Load". Which of the following describes when feedwater flow is transferred from the Startup Feedwater Pump to a Main Feedwater Pump, and why?

A. Prior to 3% Rated Thermal Power (RTP) to prevent exceeding the capacity of the Startup Feedwater Pump.

B. After exceeding 3% Rated Thermal Power (RTP) because experience has shown that this power level coincides with the minimum controllable feedwater flow through the Feedwater Regulating Bypass Valves.

C. Prior to 5% Rated Thermal Power (RTP) because the Emergency Feedwater System must be declared operable prior to entering MODE 1.

D. After exceeding 5% Rated Thermal Power (RTP) to ensure adequate feedwater flow to prevent thermal stratification of feedwater entering the Steam Generators.

Proposed Answer: A

Explanation (Optional):

A is correct. Per OS1000.02, "Plant Startup from Hot Standby to Minimum Load" there is a caution statement which states "Do NOT exceed the capacity of the Startup Feed Pump (191 amps or 3% RTP)".

B is incorrect but plausible. Operation of the feedwater control station at low power levels, including operation of the bypass valves and transfer to/from the bypass and main feedwater regulating valves has been historically challenging for operators at the feedwater station, however this is not the reason for transferring from the SUFP.

C is incorrect but plausible because the SUFP is part of the Emergency Feedwater System, however this is not the reason for transferring from the SUFP.

<table>
<tr><td colspan="6">D is incorrect but plausible. Thermal transient concerns with low feedwater flow to the steam generators is a concern, however this is not the reason for transferring from the SUFP.</td></tr>
<tr><td colspan="2">Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number)</td><td colspan="4">• OS1000.02, "Plant Startup from Hot Standby to Minimum Load", Caution Statement prior to Step 4.11.</td></tr>
<tr><td colspan="4">Proposed references to be provided to applicants during examination:</td><td colspan="2">None</td></tr>
<tr><td>Learning Objective:</td><td colspan="5"></td></tr>
<tr><td>Question Source:</td><td>Bank #</td><td>X</td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>Modified Bank#</td><td></td><td></td><td colspan="2">(Note changes or attach Parent)</td></tr>
<tr><td></td><td>New</td><td></td><td></td><td colspan="2"></td></tr>
<tr><td>Question History:</td><td>Last NRC Exam</td><td colspan="4">Indian Point Unit 3 2010</td></tr>
<tr><td rowspan="2">Question Cognitive Level:</td><td colspan="3">Memory or Fundamental Knowledge</td><td>X</td><td></td></tr>
<tr><td colspan="3">Comprehension or Analysis</td><td></td><td></td></tr>
<tr><td rowspan="2">10 CFR Part 55 Content:</td><td>55.41</td><td colspan="4">41.5</td></tr>
<tr><td>55.43</td><td colspan="4"></td></tr>
<tr><td colspan="6">Comments: The question was editorialized to be Seabrook Station specific, however the question does not meet the threshold for "significantly modified" from the version appearing on the Indian Point Unit 3 2010 NRC Exam.</td></tr>
</table>

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#45** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 061 (SF4S AFW) Auxiliary/Emergency Feedwater<br>061K6.02 - Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps | |
| | Importance Rating | 2.6 | |

Proposed Question:

Given the following conditions:

- The plant was at 100% power when a reactor trip and Safety Injection (SI) signal occurred.
- The crew is implementing E-0, "Reactor Trip or Safety Injection".
- After the crew completes the E-0 immediate actions the BOP Control Board Operator notes the following conditions:
  - Emergency Feedwater (EFW) Flow is 0 gpm.
  - FW-P-37B, "Electric Driven EFW Pump" is running with very low amps.
  - Steam supply valves to FW-P-37A, "Steam Driven EFW Pump" are in the following configuration:
    - MS-V-393, "SG A Steam Supply to EFW Pump" is OPEN
    - MS-V-394, "SG B Steam Supply to EFW Pump" is OPEN
    - MS-V-395, "Main Steam to Emergency Feedwater Pump" is CLOSED

What is the status of the Emergency Feedwater Pumps and what procedural action should the crew take to establish EFW flow?

A. FW-P-37B, "Electric Driven EFW Pump" has a sheared shaft. MS-V-395, "Main Steam to Emergency Feedwater Pump" should have opened. The crew should OPEN MS-V-395 per E-0, "Attachment A", step 11 ("Verify EFW Flow Greater than 500 GPM").

B. FW-P-37A, "Steam Driven EFW Pump" has tripped on overspeed. FW-P-37B, "Electric Driven EFW Pump" is operating at runout condition. The crew should start the Startup Feedwater Pump per E-0, Attachment T, "Establishing EFW SUFP Flow".

C. FW-P-37A, "Steam Driven EFW Pump" has tripped on overspeed. FW-P-37B, "Electric

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| | Driven EFW Pump" is operating at runout condition. The crew should reset MS-V-395, "Main Steam to Emergency Feedwater Pump" per E-0"Attachment A", step 11 ("Verify EFW Flow Greater than 500 GPM"). |
| D. | FW-P-37B, "Electric Driven EFW Pump" has a sheared shaft. FW-P-37A, "Steam Driven EFW Pump" has tripped on overspeed. The crew should start the Startup Feedwater Pump per E-0, Attachment T, "Establishing EFW SUFP Flow". |

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| Proposed Answer: | A | |

Explanation (Optional):

A is correct. The combinations of conditions in the question stem indicate that FW-P-37B has a sheared shaft. MS-V-395 should have opened 28 seconds after either MS-V-393 or 394 opened. E-0, "Attachment A", step 11 ("Verify EFW Flow Greater than 500 GPM") contains guidance to align valves as necessary to establish flow.

B is incorrect but plausible. FW-P-37A does have pump overspeed protection, however that is provided by MS-V-129, "Emergency Feed Pump Turbine Trip Valve" and not MS-V-395. Additionally, pump runout conditions on pump 37B may be possible if there were a catastrophic break in its discharge line. Also, E-0 Attachment A does provide guidance for starting the SUFP, however FW-P-37A is available and MS-V-395 should be opened per Attachment A, Step 11.

C is incorrect but plausible. FW-P-37A does have pump overspeed protection, however that is provided by MS-V-129, "Emergency Feed Pump Turbine Trip Valve" and not MS-V-395. Additionally, pump runout conditions on pump 37B may be possible if there were a catastrophic break in its discharge line. Also, E-0, Attachment A does provide guidance for resetting the steam supply to FW-P-37A, however the overspeed protection provided by MS-V-129, "Emergency Feed Pump Turbine Trip Valve" and that valve is addressed in Attachment A.

D is incorrect but plausible. The combinations of conditions in the question stem indicate that FW-P-37B has a sheared shaft. FW-P-37A does have pump overspeed protection, however that is provided by MS-V-129, "Emergency Feed Pump Turbine Trip Valve" and not MS-V-395. Also E-0, Attachment T, "Establishing EFW SUFP Flow" does provide guidance for establishing feed flow with the SUFP, however FW-P-37A is available and MS-V-395 should be opened per Attachment A, Step 11.

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • E-0, "Reactor Trip or Safety Injection |

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| Proposed references to be provided to applicants during examination: | None |

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| Learning | L8045I02, L8045I04, L1202I02 |

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|---|---|---|---|---|
| Objective: | | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#46** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 062 (SF6 ED AC) AC Electrical Distribution<br>062K3.03 - Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: DC system | |
| | Importance Rating | 3.7 | |

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- Non-Vital Battery ED-B-12A is removed from service for maintenance.
- MCC-523 de-energizes.

How is plant operation affected and what action should be taken?

A. Pressurizer Control Group heaters will de-energize. Per OS1201.06, "PZR Pressure Instrument/Component Failure", Step 5 ("Align Pressurizer Pressure Control") the crew should energize backup heaters as necessary.

B. A letdown isolation will occur. The crew should establish Excess Letdown per OS1202.01, "Loss of Letdown", Step 7 ("Establish Excess Letdown") and remove the normal letdown flow input to the plant calometric.

C. Steam dump capability on the "Plant Trip" controller is lost. Per ON1248.02, "Loss of Non-Vital DC Bus", Step 5 ("Verify Proper Steam Dump Operation") the crew should place steam dumps in the steam pressure control mode.

D. Main Turbine seal oil pressure will decrease. Per ON1248.02, "Loss of Non-Vital DC Bus", Step 10 ("Check Turbine Generator Operation") the crew should trip the reactor , go to E-0, "Reactor Trip or Safety Injection, when the immediate actions are complete vent the turbine generator to atmosphere.

| Proposed Answer: | D |
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Explanation (Optional):

D is correct. There is no cross tie capability between non vital DC busses 12A and 12B. If Non Vital Battery ED-B-12A is removed from service there is no backup DC supply to that bus. MCC-523 is the AC supply to the 12A DC bus battery charger. A loss of MCC-523 will

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| result in a loss of DC bus 12A, which results in a loss of the Turbine Generator Emergency Seal Oil Pump. Concurrently, a loss of MCC-523 results in a loss of the Turbine Generator Main Seal Oil Pump. A loss of both Seal Oil Pumps will result in degradation of seal oil pressure down to the generator internal hydrogen gas pressure. ON1248.02, "Loss of Non-Vital DC Bus", Step 10 "Response No Obtained" action is to trip the reactor, go to E-0, "Reactor Trip or Safety Injection, when the immediate actions are complete vent the turbine generator to atmosphere.<br><br>A is incorrect but plausible. The Non-Vital DC system does interface with pressurizer heaters, however it is associated with DC Bus 12B and would result on a loss of Group C and D Backup Heaters.<br><br>B is incorrect but plausible. The Non-Vital DC system does interface with the letdown portion of the CVCS system and would result in a loss of letdown, however it is associated with a loss of DC Bus 12B.<br><br>C is incorrect but plausible. The Non-Vital DC system does interface with the Steam Dumps and would result in a loss of Steam Dump capability, however it is associated with a loss of DC Bus 12B. |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ON1248.02, "Loss of Non-Vital DC Bus"<br>• OS1201.06, "PZR Pressure Instrument/Component Failure"<br>• OS1202.01, "Loss of Letdown" |

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|---|---|
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L8052I, L8052I08, L1189I07 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | X | Seabrook TEB#25122 | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 | 55.41 | **41.7** | |

| | 55.43 | |
|---|---|---|

Comments:

Parent question:

Plant Conditions:

- 65% power.
- Battery ED-B-12A removed from service for maintenance
- MCC 523 deenergizes
- Turbine Runback occurs.
- US enters the appropriate Abnormal Operating Procedure

How is plant operation affected and what direction should the US provide?

A. Reactor will trip on high pressurizer pressure. Manually trip reactor.
B. Reactor will trip on lo-lo Steam Generator level. Manually trip reactor.
C. Seam dumps will not open and SG pressures will increase to > 1150 psig. Manually trip reactor.
D. Seal oil pressure will decrease to Main Turbine supply pressure. Manually trip the reactor and local vent the generator to atmosphere.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#47** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 063 (SF6 ED DC) DC Electrical Distribution<br><br>063A1.01 - Ability to predict and/or monitor changes in parameters associated with operating the DC electrical system controls including: Battery capacity as it is affected by discharge rate | |
| | Importance Rating | 2.5 | |

Proposed Question:

A battery is supplying its Vital 125VDC bus. Which of the following would be an indication in the Control Room that a heavy load has been placed on the battery?

A. No initial change in voltage but a decrease over time.

B. An initial drop in voltage with a slow return to normal.

C. An initial drop in voltage with a gradual decrease over time.

D. An initial increase in voltage with a gradual decrease over time.

Proposed Answer: C

Explanation (Optional):

C is correct. As a battery cell discharges tis voltage will begin to drop. On a loss of all AC power, when the battery has a high load, the voltage drops at a rapid rate immediately, then at a slower rate as time progresses. Voltage is indicative of the batteries condition under load, so the Control Room crew can use voltage as an approximation of battery capacity.

All of the distractors are incorrect but plausible if the student has a misconception of the relationship between voltage and load.

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| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • ECA-0.0, Step 16 note |
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | | | | |
| Question Source: | Bank # | X | Diablo Canyon 2010 NRC Exam | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.5** | | |
| | 55.43 | | | |
| | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#48** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 063 (SF6 ED DC) DC Electrical Distribution<br>063A4.01 - Ability to manually operate and/or monitor in the control room: Major breakers and control power fuses | |
| | Importance Rating | 2.8 | |

Proposed Question:

Given the following conditions:

- The plant was initially at 100% power.
- A Safety Injection occurred.
- The Emergency Core Cooling System is running as expected.
- Subsequently, a fault in the 125VDC system results in a loss of DC Bus 11A.

Which of the following correctly describes the impact of this condition:

A. The "A" Emergency Diesel Generator output breaker opens and cannot be operated from the Main Control Board. The Train "A" Charging Pump, Safety Injection Pump, and RHR Pump breakers all open and can be operated from the Main Control Board.

B. The "A" Emergency Diesel Generator output breaker remains open and cannot be operated from the Main Control Board. The Train "A" Charging Pump, Safety Injection Pump, and RHR Pump breakers remain closed and cannot be operated from the Main Control Board.

C. The "A" Emergency Diesel Generator output breaker remains closed and can be operated from the Main Control Board if necessary.. The Train "A" Charging Pump, Safety Injection Pump, and RHR Pump breakers all open and cannot be operated from the Main Control Board.

D. The "A" Emergency Diesel Generator output breaker remains open and can be operated from the Main Control Board if necessary. The Train "A" Charging Pump, Safety Injection Pump, and RHR Pump breakers remain closed and can be operated from the Main Control Board if needed.

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| Proposed Answer: | B | |
| Explanation (Optional): | | |

B is correct. Given the conditions in the question stem the "A" Emergency Diesel Generator would be running due to the SI signal, however the output breaker remains open on an SI. With a loss of DC Bus 11A there is no control power to the DG Output Breaker, "A" Charging Pump Breaker, "A" SI Pump , and "A" RHR Pump. With no control power those breakers cannot be operated from the MCB.

A is incorrect but plausible. The student may incorrectly determine that the DG Output Breaker would fail open on a loss of control power. Additionally, the student may believe that safety related ECCS pumps have a backup DC control power supply and are still capable of being operated remotely.

C is incorrect but plausible. The student may incorrectly determine that the DG Output Breaker has a backup DC control power supply and is still capable of being operated remotely. Additionally, it is true that with a loss of DC Bus 11A there is no control power to the "A" Charging Pump Breaker, "A" SI Pump , and "A" RHR Pump, however with no control power those breakers cannot be operated from the MCB.

D is incorrect but plausible. It is true that the DG Output Breaker would remain open, however, however with no control power it cannot be operated from the MCB. Additionally, it is true that the "A" Charging Pump Breaker, "A" SI Pump , and "A" RHR Pump breakers would remain closed, however with no control power those breakers cannot be operated from the MCB.

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1248.01, "Loss of a Vital DC Bus"<br>• Drawing 310102, Sheet A54c, "DG "A" Breaker Close Schematic<br>• Drawing 310102, Sheet A54d, "DG "A" Breaker Trip Schematic<br>• Drawing 310887, Sheet A57b,"A" RHR Pump Breaker Close Schematic<br>• Drawing 310887, Sheet A57c,"A" RHR Pump Breaker Trip Schematic<br>• Drawing 310890, Sheet A56b,"A" SI Pump Breaker Close Schematic<br>• Drawing 310890, Sheet A56c,"A" SI Pump Breaker Trip Schematic<br>• Drawing 310891, Sheet A62b,"A" Charging Pump Breaker Close Schematic<br>• Drawing 310891, Sheet A62c,"A" Charging Pump Breaker Trip Schematic |

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|---|---|---|---|---|---|
| Proposed references to be provided to applicants during examination: | | | | | None |
| Learning Objective: | L8017I01, L1189I04 | | | | |
| Question Source: | Bank # | X | Seabrook TEB #26728 | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | | | | |
| Question History: | Last NRC Exam | | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | | |
| | Comprehension or Analysis | X | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | | |
| | 55.43 | | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#49** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 064 (SF6 EDG) Emergency Diesel Generator<br>064A1.04 - Ability to predict and/or monitor changes in parameters(to prevent exceeding design limits) associated with operating the ED/G system controls including: Crankcase temperature and pressure | |
| | Importance Rating | 2.8 | |

| Proposed Question: | |
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Given the following conditions:

- The "A" Emergency Diesel Generator was started for surveillance testing and has been running for 10 minutes.
- The NSO at the diesel generator control panel informs the control room that local alarm "Crankcase Pressure High" is in alarm AND that the Crankcase Exhauster Fan is NOT running.
- The NSO reports that crankcase pressure is .65" of water and slowly rising.

What action should the crew take and why?

A. Per the "Crankcase Pressure High" alarm response procedure the control room crew should direct the NSO to start the Crankcase Exhauster Fan because it should have started when crankcase pressure reached .6" of water.

B. Per the "Crankcase Pressure High" alarm response procedure the control room crew should unload and stop the diesel because the diesel engine will automatically shut down when crankcase pressure exceeds .7" of water.

C. Per the "Crankcase Pressure High" alarm response procedure the control room crew should direct the NSO to start the Crankcase Exhauster Fan because it should have started when the diesel engine low speed relay actuated upon engine startup.

D. Per the "Crankcase Pressure High" alarm response procedure the control room crew should stop the diesel using the Emergency Stop Pushbuttons because the diesel engine should have automatically shut down when crankcase pressure exceeded .6" of water.

| Proposed Answer: | C | |
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| Explanation (Optional): | | |

C is correct. The crankcase exhauster fan should automatically start when the engines low speed relay actuates at 125rpm.

A is incorrect but plausible. It is true that the local alarm response procedure (UA-9558 C-1) directs the operator to start the crankcase exhauster fan, however the signal for the fan to auto start is from the engines low speed relay.

B is incorrect but plausible. It is plausible that the procedure would direct the operators to stop the diesel as a crankcase explosion may well be preceded by an increasing crankcase pressure and an engine automatic trip signal, however there is no automatic engine shutdown signal based on crankcase pressure.

D is incorrect but plausible. It is plausible that the procedure would direct the operators to stop the diesel as a crankcase explosion may well be preceded by an increasing crankcase pressure and an engine automatic trip signal, however there is no automatic engine shutdown signal based on crankcase pressure. Furthermore, the setpoint of 6" of water is valid, however it is associated with the "Crankcase Pressure High" alarm setpoint.

| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • Local Alarm Response Procedure UA-9558 C-1, "Crankcase Pressure High" |
|---|---|
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L8019I06 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | **41.5** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#50** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 073 (SF7 PRM) Process Radiation Monitoring<br>073K3.01 - Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases | |
| | Importance Rating | 3.6 | |

Proposed Question:

Given the following plant conditions:

- Containment Online Purge (COP) is in service.
- COP-RM-6527A1, "COP Train A Rad Monitor Channel A1" malfunctions and goes into HIGH alarm.
- COP-RM-6527A2, "COP Train A Rad Monitor Channel A2" is functioning properly and is not in alarm.
- COP-RM-6527B1, "COP Train A Rad Monitor Channel B1" is functioning properly and is not in alarm.
- COP-RM-6527B2, "COP Train A Rad Monitor Channel B2" is functioning properly and is not in alarm.
- All other equipment functions properly.

Which of the following describes the status of the CVI (Containment Ventilation Isolation) signal actuation?

A. The CVI signal is actuated.

B. The CVI signal will actuate if COP-RM-6527A2 also goes into HIGH alarm.

C. The CVI signal will actuate if COP-RM-6527B1 also goes into HIGH alarm.

D. The CVI signal will actuate if COP-RM-6527B2 also goes into HIGH alarm.

| Proposed Answer: | B |
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Explanation (Optional):

B is correct. A CVI signal is generated when 2/2 detectors on either train go into HIGH

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| alarm. With COP-RM-6527A1 failed to HIGH alarm CVI would actuate if the second detector from the same train, COP-RM-6527A2 goes into HIGH alarm.<br><br>Distractors A, C, and D are all plausible as the students must know that the CVI signal coincidence is 2/2 detectors on 1/2 trains. All three distractors are associated with a misconception of the CVI signal logic. | | | | | |
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1252.02, Airborne High Radiation, Step 2 | | | | |
| Proposed references to be provided to applicants during examination: | | | | None | |
| Learning Objective: | L8038I23 | | | | |
| Question Source: | Bank # | | | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | X | | | |
| Question History: | Last NRC Exam | | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | X | |
| | Comprehension or Analysis | | | | |
| 10 CFR Part 55 Content: | 55.41 | 41.7 | | | |
| | 55.43 | | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#51** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 073 Process Radiation Monitoring (PRM) System<br>K4.01 Knowledge of the PRM system design feature(s) and/or interlock(s) which provide for the following: Release termination when radiation exceeds setpoint. | |
| | Importance Rating | 4.0 | |

Proposed Question:

Given the following conditions:

- The Steam Generator Blowdown Flash Tank discharge is aligned to the ocean for a demineralizer conductivity flush.
- Blowdown flow from each Steam Generator is 20 gallons per minute.
- RM6510-1, Steam Generator A Blowdown Radiation Monitor goes into Hi Alarm.

NOTE:

IRC= Inside Reactor Containment Isolation Valves.

ORC= Outside Reactor Containment Isolation Valves.

How will the IRC and ORC Containment Isolation Valves respond?

A. The Steam Generator IRC valves will all immediately close.

B. Only the Steam Generator "A" IRC valve will immediately close.

C. SB-CV-6519, "SG Flash Tank Discharge Valve" will immediately close and the IRC valves will all close when the Flash Tank High Level setpoint is reached.

D. SB-CV-6519, "SG Flash Tank Discharge Valve" will immediately close and the ORC valves will all close when the Flash Tank High Level setpoint is reached.

Proposed Answer: C

Explanation (Optional):

C is correct. A high radiation alarm on any of the individual SB lines will cause SB-CV-6519 to close. This will cause the SB Flash Tank level to continuously increase until the

high level alarm is reached. The IRC valves will automatically close on SB high level.

A is incorrect but plausible. The IRC valves will close however it is not immediate. A high alarm on any of the individual SB lines will cause SB-CV-6519 to close. This will cause the SB Flash Tank level to continuously increase until the high level alarm is reached. The IRC valves will automatically close on SB high level.

B is incorrect but plausible. It is true that the “A” IRC valve will close, however it is not immediate and the “B”, “C”, and “D” IRC valves will also close.

D is incorrect but plausible. It is true that a high radiation alarm on any of the individual SB lines will cause SB-CV-6519 to close. This will cause the SB Flash Tank level to continuously increase until the high level alarm is reached., however it is the IRC valves that will automatically close on SB high level.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1252.01, “Process or Effluent High Radiation”, Attachment A: Process and Effluent Radiation Monitors |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L8063I11 |

| | | | | |
|---|---|---|---|---|
| Question Source: | Bank # | | | |
| | Modified Bank# | X | Seabrook TEB#22601 | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 | 55.41 | **41.7** | |

| | 55.43 | |
|---|---|---|

Comments:

Parent Question:

The following conditions exist:

- 100% power.
- RM-6510, STM GEN “A” Blowdown Radiation Monitor goes into Hi alarm.
- The SB system is in a normal operating lineup.

NOTE:
IRC= Inside Reactor Containment Isolation Valves.
ORC= Outside Reactor Containment Isolation Valves.

How does the SB Flash Tank response?

A. Level will increase until the IRC valves close on high flash tank level.

B. Level will increase until the ORC valves close on high flash tank level.

C. Level will increase until the IRC AND ORC valves close on high flash tank level.

D. Pressure will increase until the IRC valves close on high flash tank pressure.

Correct answer: A

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#52** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 076 Service Water System (SWS)<br>K2.08 Knowledge of the bus power supplies to the following: ESF-actuated MOVs | |
| | Importance Rating | 3.1 | |

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 |
|---|---|

Explanation (Optional):

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.

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.

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):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1016.05, Service Water Cooling Tower Operation<br>• 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 |

| | | | | |
|---|---|---|---|---|
| Learning Objective: | L8037I06 | | | |
| Question Source: | Bank # | X | Seabrook 2009 NRC Remediation Exam | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |

| | | |
|---|---|---|
| 10 CFR Part 55 Content: | 55.41 | **41.7** |
| | 55.43 | |
| Comments: | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#53** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 076 (SF4S SW) Service Water<br>076K4.01 - Knowledge of SWS design feature(s) and/or interlock(s)which provide for the following: Conditions initiating automatic closure of closed cooling water auxiliary building header supply and return valves | |
| | Importance Rating | 2.5 | |

Proposed Question:

Given the following plant conditions:

- The plant was initially at 100% power.
- Both trains of Service Water were aligned to the Cooling Tower per the normal operating procedure.
- Steam Generator "A" completely depressurizes.
- The plant responds as designed.
- The crew has isolated the faulted SG.
- Secondary radiation is normal.
- The crew has verified all AC busses energized from the UATs per the applicable EOP.

What caused SW-V-5, SW Isolation to Secondary Loads, to close?

A. Emergency power sequencer relay PR1.

B. Tower Actuation.

C. Safety Injection Actuation.

D. Reactor Trip P-4 signal.

Proposed Answer: C

Explanation (Optional):

C is correct. SW-V-5 is the Train 'B' SW isolation valve for secondary/turbine building loads. The valve will automatically close on either a) Emergency Power Sequencer relay PR1, b) Tower Actuation Signal (low SW system pressure w/ocean SW pump running for

>28 seconds) or c) a Safety Injection signal. The stem of the question indicates that a steamline break occurred and that the crew took procedural actions to isolate a faulted steam generator. These conditions indicate that a Safety Injection signal would have actuated.

A is incorrect but plausible. If there is a loss of electrical power and the Emergency Power Sequencer actuates then the PR1 contact would cause SW-V-5 to automatically close however the question stem states that the crew verified emergency bus power from the UAT's so a loss of power and EPS actuation would not have occurred.

B is incorrect but plausible. A Tower Actuation signal will cause SW-V-5 to close, however that signal would not have been generated because the SW system was previously aligned to the cooling tower. With the SW system aligned to the cooling tower the ocean SW pump breakers would not be closed so a Cooling Tower actuation signal due to low SW system pressure would not occur. Additionally, the question stem states that the crew verified emergency bus power from the UAT's so a Tower Actuation signal from a loss of power condition would not have occurred.

D is incorrect but plausible. It is plausible that SW to the Turbine Building would isolate automatically on a reactor trip to support heat removal from primary loads as is the case on a Safety Injection signal however a reactor trip P-4 signal does not feed into the automatic close logic for SW-V-5.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-503962, SW-Cooling Tower Actuation Logic Diagram, Sheet 1 of 2<br>• 1-NHY-503977, SW-Isolation Valves For SW Flow to Turbine Building Logic Diagram<br>• 1-NHY-503979, SW-Cooling Tower Actuation Logic Diagram, Sheet 2 of 2 |

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |

| | | | | |
|---|---|---|---|---|
| Learning Objective: | L8037I06 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2010 NRC Exam | | |

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

| | | |
|---|---|---|
| | 55.43 | |
| Comments: | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#54** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 078 (SF8 IAS) Instrument Air<br>078G2.4.47 - Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. | |
| | Importance Rating | 4.2 | |

Proposed Question:

**NOTE: This is an open reference question.**

Given the following plant conditions:

- The plant is at 100% power.
- You are the BOP control room operator.
- At 2335 a loss of main plant computer occurs immediately followed by a decreasing instrument air pressure trend.
- Your crew has entered ON1242.01, "Loss of Instrument Air" AND ON1251.01, "Loss of Plant Computer.
- Per ON1251.01, "Loss of Plant Computer" your crew has been recording local area temperature readings on Attachment C, the local temperature reading log sheet.
- Per ON1242.01, "Loss of Instrument Air", Step 7, "Initiate Area Temperature Monitoring", at 2345 the Unit Supervisor directed you to implement Attachment G, "Area Temperature Monitoring Actions".
- The time is now 0200.

Utilizing ON1242.01, Attachment G, "Area Temperature Monitoring Actions", which of the following correctly describes the required action or actions that must be taken?

A. Go to ON1223.01, "Loss of Control Room Ventilation or Air Conditioning" ONLY.

B. Go to the VAS alarm hard copy for B6543, "Battery Room A Temp High" ONLY.

C. Go to ON1223.01, "Loss of Control Room Ventilation or Air Conditioning" AND go to the VAS alarm hard copy for B6543, "Battery Room A Temp High".

D. Continue to monitor area temperature VAS alarms for the Control Room , Essential Switchgear (Train A and B), and Vital Battery Rooms A, B, C, and D. If any of the area

| |
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| temperature alarms actuates then perform the actions of the associated VAS Alarm Response Procedure. |

| | | |
|---|---|---|
| Proposed Answer: | C | |
| Explanation (Optional): | | |

C is correct. Per ON1242.01, Attachment G, "Area Temperature Monitoring Actions , if the MPCS is not available then the action is to perform ON1251.01, Attachment D, "Area Temperature Monitoring Actions", which in turn directs implementing attachment C, the area temperature log sheet. At the 0200 the data on the log sheet indicates that the Control Room area and Battery Room "A" temperatures exceed the prescribed limits. In this case Attachment D directs the operator to go to ON1223.01, "Loss of Control Room Ventilation or Air Conditioning" AND go to the VAS alarm hard copy for B6543, "Battery Room A Temp High".

A is incorrect but plausible. It is true that ON1251.01, "Loss of Plant Computer, Attachment D would direct the operator to go to ON1223.01, "Loss of Control Room Ventilation or Air Conditioning", however the operator should also go to the VAS alarm hard copy for B6543, "Battery Room A Temp High".

B is incorrect but plausible. It is true that ON1251.01, "Loss of Plant Computer, Attachment D would direct the operator to go to the VAS alarm hard copy for B6543, "Battery Room A Temp High", however the operator should also go to ON1223.01, "Loss of Control Room Ventilation or Air Conditioning".

D is incorrect but plausible. This distractor describes the action that would be taken per ON1242.01, Attachment G, "Area Temperature Monitoring Actions if the MPCS were available.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ON1242.01, Attachment G, "Area Temperature Monitoring Actions"<br>• ON1251.01, "Loss of Plant Computer, Attachment D, "Area Temperature Monitoring Actions" |
| Proposed references to be provided to applicants during examination: | • ON1242.01, Attachment G, "Area Temperature Monitoring Actions"<br>• ON1251.01, "Loss of Plant Computer, Attachment C, Local area temperature reading log sheet. **(With data filled in)**<br>• ON1251.01, "Loss of Plant Computer, Attachment D, "Area Temperature Monitoring Actions" |
| Learning Objective: | L1194I02 |

| | | | | |
|---|---|---|---|---|
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | **41.10** | | |

| | | |
|---|---|---|
| | 55.43 | **43.5** |

Comments:

- The document shown below is the open reference ON1251.01, "Loss of Plant Computer, Attachment C, Local area temperature reading log sheet. **(With data filled in)**

## ATTACHMENT C

(Page 1 of 4)

MAIN PLANT COMPUTER OUT OF SERVICE [illegible] DATE/TIME RETURNED ______________

| PARAMETER | LIMITS | | | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | 0000 | 0100 | 0200 | 0300 | 0400 | 0500 | 0600 | 0700 | 0800 | 0900 | 1000 | 1100 | 1200 | 1300 | 1400 | 1500 | 1600 | 1700 | 1800 | 1900 | 2000 | 2100 | 2200 | 2300 |
| | | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] | [illegible] |
| CONTROL ROOM AREA | [illegible]°F | 71 | 73 | 76 | | | | | | | | | | | | | | | | | | | | | |
| | | | | | | | | | | | | | | | | | | | | | | | | | |
| TRAIN A SWGR AREA | 95.5°F | 70 | 71 | 74 | | | | | | | | | | | | | | | | | | | | | |
| | 50°F | | | | | | | | | | | | | | | | | | | | | | | | |
| TRAIN B SWGR AREA | 95.5°F | 71 | 73 | 72 | | | | | | | | | | | | | | | | | | | | | |
| | 50°F | | | | | | | | | | | | | | | | | | | | | | | | |
| BATTERY ROOM A | 90.5°F | 82 | 86 | 93 | | | | | | | | | | | | | | | | | | | | | |
| | 65°F | | | | | | | | | | | | | | | | | | | | | | | | |
| BATTERY ROOM B | 90.5°F | 78 76 | 81 | 84 | | | | | | | | | | | | | | | | | | | | | |
| | 65°F | | | | | | | | | | | | | | | | | | | | | | | | |
| BATTERY ROOM C | 90.5°F | 76 | 77 | 76 | | | | | | | | | | | | | | | | | | | | | |
| | 65°F | | | | | | | | | | | | | | | | | | | | | | | | |
| BATTERY ROOM D | 90.5°F | 75 | 74 | 76 | | | | | | | | | | | | | | | | | | | | | |
| | 65°F | | | | | | | | | | | | | | | | | | | | | | | | |

Perform actions required by ATTACHMENT D for affected areas whose temperatures approach or exceed limits.

Perform [illegible] temperature monitoring as follows:

A Train - Switchgear and Battery Rooms – Every 4 Hours
B Train - Switchgear and Battery Rooms – Every 12 Hours
Control Room Area – Every 30 minutes

ON1251.01-C0
Rev 00

ATTACHMENT C CONTINUED ON THE NEXT PAGE

## ATTACHMENT C

(Page 2 of 4)

DATE/TIME INITIATED: ______________

| PARAMETER | UNITS | 0000 | 0100 | 0200 | 0300 | 0400 | 0500 | 0600 | 0700 | 0800 | 0900 | 1000 | 1100 | 1200 | 1300 | 1400 | 1500 | 1600 | 1700 | 1800 | 1900 | 2000 | 2100 | 2200 | 2300 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| CIRC WATER DISCH TEMP [illegible] | NA | 78 | 78 | 78 | | | | | | | | | | | | | | | | | | | | | |
| CIRC WATER INTAKE TEMP [illegible] | NA | 50 | 50 | 50 | | | | | | | | | | | | | | | | | | | | | |
| CIRC WATER ΔT (CW-TI-5983) [illegible] | °F | 28 | 28 | 28 | | | | | | | | | | | | | | | | | | | | | |
| SEISMIC MONITORING PANEL (Note 1) | SAT | SM | SM | SM | | | | | | | | | | | | | | | | | | | | | |

TIME FREQUENCY: HOURLY

TIME FREQUENCY – 4 HOURS

| PARAMETER | UNITS | | | | | | |
|---|---|---|---|---|---|---|---|
| CONTROL ROD POSITION INDICATOR | DRPI/DEMAND WITHIN 12 STEPS ALL RODS WITHIN RIL | 0000 | 0100 | 0200 | | | |
| CONTROL BANK A | 230 | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT |
| CONTROL BANK B | 230 | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT |
| CONTROL BANK C | 230 | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT |
| CONTROL BANK D | 230 | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT |
| SHUTDOWN BANK A | 230 | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT |
| SHUTDOWN BANK B | 230 | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT |
| SHUTDOWN BANK C | 230 | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT |
| SHUTDOWN BANK D | 230 | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT |
| SHUTDOWN BANK E | 230 | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT | SAT / UNSAT |

* Only required if CW-TI-5981 is not operating

Note 1: Graphical User Interface indicators Green and no alarm lit on alarm panel.

ATTACHMENT C CONTINUED ON THE NEXT PAGE

**ATTACHMENT C**

(Page 3 of 4)

| PARAMETER | LIMITS | [illegible] |
|---|---|---|
| [illegible] | [illegible] | [illegible] |

| PARAMETER | LIMITS | [illegible] |
|---|---|---|
| [illegible] | [illegible] | [illegible] |

| PARAMETER | LIMITS | [illegible] |
|---|---|---|
| [illegible] | [illegible] | [illegible] |

| PARAMETER | LIMITS | [illegible] |
|---|---|---|
| [illegible] | [illegible] | [illegible] |

**ATTACHMENT C CONTINUED ON THE NEXT PAGE**

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#55** | Tier # | 2 | |
| | Group # | 1 | |
| | K/A # | 103 (SF5 CNT) Containment<br>103K4.04 - Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following: Personnel access hatch and emergency access hatch | |
| | Importance Rating | 2.5 | |

Proposed Question:

Operability of the Containment Personnel Hatch Air Lock is determined by a surveillance performed at the local test panel. The hatch test panel is designed to test operability based on_______________.

A. air leakage rate across the door seals

B. differential pressure across the door seals

C. rate of change of differential pressure across the door seals

D. proper limit switch indication for the mechanical door latches

Proposed Answer: A

Explanation (Optional):

A is correct. Per OX1460.01, “Weekly Air Lock Door Seal Air Flow Rate” the inner and outer door seals are tested and verified operable utilizing the air flow rate value across the inner and outer door seals.

B is incorrect but plausible. Differential pressure across the seals could be an indicator of seal integrity, however that parameter is not used.

C is incorrect but plausible. Rate of change of differential pressure across the seals could be an indicator of seal integrity, however that parameter is not used.

D is incorrect but plausible. It is true that the inner and outer doors have a rotating mechanical latch feature however there is no latch position testing feature.

<table>
<tr><td>Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number)</td><td colspan="5">• Technical Specification 3.9.4, Containment Building Penetrations</td></tr>
<tr><td colspan="4">Proposed references to be provided to applicants during examination:</td><td colspan="2">None</td></tr>
<tr><td>Learning Objective:</td><td colspan="5">L8010I04</td></tr>
<tr><td>Question Source:</td><td>Bank #</td><td></td><td></td><td colspan="2"></td></tr>
<tr><td></td><td>Modified Bank#</td><td></td><td></td><td colspan="2">(Note changes or attach Parent)</td></tr>
<tr><td></td><td>New</td><td>X</td><td></td><td colspan="2"></td></tr>
<tr><td>Question History:</td><td>Last NRC Exam</td><td colspan="4"></td></tr>
<tr><td rowspan="2">Question Cognitive Level:</td><td colspan="3">Memory or Fundamental Knowledge</td><td></td><td></td></tr>
<tr><td colspan="3">Comprehension or Analysis</td><td>X</td><td></td></tr>
<tr><td rowspan="2">10 CFR Part 55 Content:</td><td>55.41</td><td colspan="4">41.7</td></tr>
<tr><td>55.43</td><td colspan="4"></td></tr>
<tr><td colspan="6">Comments:</td></tr>
</table>

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#56** | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 001 (SF1 CRDS) Control Rod Drive<br>001K5.88 - Knowledge of the following operational implications as they apply to the CRDS: Effects of boron on temperature coefficient | |
| | Importance Rating | 2.9 | |

Proposed Question:

Given the following plant conditions:

- The plant is initially at 100% power.
- Core age is at End of Life.
- Control Rods are in automatic.
- Control Bank 'D' is at 228 steps.
- Subsequently ONE Control Bank 'D' rod drops to the bottom of the core.
- The Reactor does not trip.
- No operator action is taken.

What will be the Reactor Coolant System (RCS) temperature response?

A. RCS temperature will decrease and remain at a new lower value.

B. RCS temperature will initially decrease and then return to program as control rods respond to a Tavg/Rref error signal.

C. RCS temperature will initially decrease and then return to higher than initial value.

D. RCS temperature will initially decrease and then return to program due to positive reactivity from the Moderator Temperature Coefficient.

Proposed Answer: A

Explanation (Optional):

A is correct. Control Bank D is at 228 steps which is above the C-11 setpoint of 223 steps. C-11 blocks automatic control rod withdrawal above 223 steps on Bank 'D'. When the control rod drops reactor coolant temperature will decrease due to the addition of negative reactivity. With the core at End of Life there is a negative moderator temperature coefficient due to reduced boron concentration. The decrease in RCS temperature would

| |
|---|
| add positive reactivity due to the MTC but not enough to return reactor power/RCS temperature back to program.<br><br>B is incorrect but plausible. If control rods were capable of automatic withdrawal then the lowering RCS temperature and disparity between nuclear power and Tref power would call for outward rod motion. Control Bank D is at 228 steps which is above the C-11 setpoint of 223 steps. C-11 blocks automatic control rod withdrawal above 223 steps on Bank 'D'.<br><br>C is incorrect but plausible. If control rods were capable of automatic withdrawal then the lowering RCS temperature and disparity between nuclear power and Tref power would call for outward rod motion. Control Bank D is at 228 steps which is above the C-11 setpoint of 223 steps. C-11 blocks automatic control rod withdrawal above 223 steps on Bank 'D'. Temperature would not return to a value higher than original.<br><br>D is incorrect but plausible. It is true that RCS temperature would decrease however the effects of the moderator temperature coefficient would add positive reactivity at the new lower temperature. Temperature would not return to program. |

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-509049, Rod Control and Blocks w/ Functional Diagram |
| Proposed references to be provided to applicants during examination: | None |

| | | | | |
|---|---|---|---|---|
| Learning Objective: | | | | |
| Question Source: | Bank # | X | Seabrook 2010 NRC Exam | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#57** | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 011 (SF2 PZR LCS) Pressurizer Level Control<br>011K2.01 - Knowledge of bus power supplies to the following: Charging pumps | |
| | Importance Rating | 3.1 | |

Proposed Question:

Which of the following correctly lists components that would be unavailable if 13.8Kv Bus 2 is de-energized??

A. Pressurizer “Control Group” Heaters and Reactor Coolant Pump “B”.

B. CS-P-128, “Positive Displacement Charging Pump” and Reactor Coolant Pump “C”.

C. CS-P-2B, “Centrifugal Charging Pump B” and AR-P-50B, “Mechanical Vacuum Pump B”.

D. AR-P-50B, “Mechanical Vacuum Pump B” and Reactor Coolant Pump “B”.

Proposed Answer: B

Explanation (Optional):

B is correct. CS-P-128 is powered from 480VAC US-25, which is powered from 3.18Kv Bus 2. Additionally, Reactor Coolant Pumps “A” and “B” are powered from Bus 1 and “C” and “D” are powered from Bus 2

A is incorrect but plausible. It is true that the Pressurizer “Control Group” Heaters are powered from 480VAC US-23 however Reactor Coolant Pump “B” is powered from Bus 1.

C is incorrect but plausible. AR-P-50B is powered from a non-vital Unit Substation, however it is 480VAC US-14. Additionally, CS-P-2B, “Centrifugal Charging Pump B” is powered from vital emergency Bus E6.

D is incorrect but plausible. AR-P-50B is powered from a non-vital Unit Substation, however it is 480VAC US-14. Additionally, Reactor Coolant Pump “B” is powered from Bus 1.

| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • 1-NHY-310891, Sheet AH4A, Post Disp Charging Pump Three Line Diagram<br>• 1-NHY-310891, Sheet A82A, Charging Pump “B” Three Line Diagram |
|---|---|

| | | | | |
|---|---|---|---|---|
| | • 1-NHY-310882 Sheet AM4a, PZR Control Group Heaters<br>• 1-NHY-310853 Sheet AJ9a, Mech Vacuum Pump 50B<br>• 1-NHY-310882, Sheet A20, Reactor Coolant Pump “B”<br>• 1-NHY-310882, Sheet A09, Reactor Coolant Pump “C” | | | |
| Proposed references to be provided to applicants during examination: | | | | None |
| Learning Objective: | | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#58** | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 015 (SF7 NI) Nuclear Instrumentation<br>015K6.03 - Knowledge of the effect of a loss or malfunction on the following will have on the NIS: Component interconnections | |
| | Importance Rating | 2.6 | |

Proposed Question:

Given the following plant conditions:

- The plant is in Mode 2.
- Reactor power is stable at 4%.
- Power Range Channel NI-41 is undergoing maintenance with the control power fuses removed.
- Power Range Channel NI-43 fails high.

What is the effect on source range instrumentation and what operator action is required?

A. Both Source Range Instruments will remain de-energized. Enter E-0, "Reactor Trip or Safety Injection".

B. Both Source Range Instruments remain energized. Enter E-0, "Reactor Trip or Safety Injection".

C. Both Source Range Instruments will remain de-energized. Check redundant bistable Not Tripped.

D. Both Source Range Instruments remain energized. Check redundant bistable Not Tripped.

Proposed Answer: A

Explanation (Optional):

A is correct. Power range channel NI-41 has its control power fuses removed which puts it in a tripped condition. When power range channel NI-43 fails high there is now a 2 of 4 logic met for the Power Range High Flux Low Setpoint (25% power)Trip. Additionally, the 2 of 4 channel coincidence for signal P-10 (10% power) is met. The source range instruments will remain de-energized. With the reactor trip signal actuated procedure E-0, Reactor Trip or Safety Injection should be implemented.

B is incorrect but plausible. When power range channel NI-43 fails high there is now a 2 of 4 logic met for the Power Range High Flux Low Setpoint (25% power). With the reactor trip signal actuated procedure E-0, Reactor Trip or Safety Injection should be implemented. This answer is wrong however as the source range instruments would be automatically deenergized by P-10. There is a common operator misconception regarding the interface between P-10 and the source range instruments. P-10 automatically deenergizes the source range instruments vice allows for manual deenergization, as would be the case with the P-6 interlock.

C is incorrect but plausible. The source range instruments will remain de-energized. The requirement to check the redundant bistable is consistent with various other instrument failures, however the reactor will have tripped and E-0 applies.

D is incorrect but plausible. The source range instruments will remain de-energized vice energized. The requirement to check the redundant bistable is consistent with various other instrument failures, however the reactor will have tripped and E-0 applies.

| | | | | |
|---|---|---|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • E-0, Reactor Trip or Safety Injection, Part B, Symptoms or Entry Conditions<br>• E-0, Reactor Trip or Safety Injection, Attachment B, Symptoms that Require a Reactor Trip.<br>• 1-NHY-509043, NI and Manual Trip Signals w/ Functional Diagram<br>• 1-NHY-509044, NI Perm. And Blocks w/ Functional Diagram | | | |
| Proposed references to be provided to applicants during examination: | | | None | |
| Learning Objective: | L1202I01, L8030I08, L8030I09, L8030I12 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | 2010 Seabrook NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | 41.10 | | |
| | 55.43 | 43.2 | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#59** | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 017 (SF7 ITM) In-Core Temperature Monitor<br>017K3.01 - Knowledge of the effect that a loss or malfunction of the ITM system will have on the following: Natural circulation indications | |
| | Importance Rating | | |

Proposed Question:

Given the following conditions:

- The plant was at 100% power.
- A loss of offsite power occurred.
- The crew is performing ES-0.1, "Reactor Trip Response", Step 18, "Check RCP 1C-Running.
- The Unit Supervisor directs the primary side control board operator to verify that natural circulation has been established.
- The board operator identifies that both trains of core exit thermocouple indications are malfunctioning.

Which one of the following correctly lists available control board instruments that are used to determine that natural circulation has been established?

A. SG Pressure Instruments-Pressures Increasing, RCS Hot Leg Temperature Instruments-Temperatures Stable or Decreasing, and RCS Cold Leg Temperature Instruments-Temperatures at Saturation Temperature For SG Pressure

B. SG Pressure Instruments-Pressures Stable or Decreasing, RCS Hot Leg Temperatures Instruments- Temperatures at Saturation Temperature For SG Pressure, and RCS Cold Leg Temperature Instruments- Temperatures Stable or Decreasing

C. SG Pressure Instruments-Pressures Stable or Decreasing, RCS Hot Leg Temperature Instruments-Temperatures Stable or Decreasing, and RCS Cold Leg Temperatures Instruments-Temperatures at Saturation Temperature For SG Pressure

D. RCS Subcooling Instruments-Subcooling Greater than 40°F, SG Pressure Instruments-Stable or Decreasing, RCS Hot Leg Temperature Instruments-Stable or Decreasing,

| |
|---|
| and RCS Cold Leg Temperatures-At Saturation Temperature For SG Pressure |

| | | |
|---|---|---|
| Proposed Answer: | C | |

Explanation (Optional):

C is correct. Per OS-0.1, Attachment I, the following conditions indicate natural circulation flow:

- RCS Subcooling- Greater than 40°F
- SG Pressures-Stable or Decreasing
- RCS Hot Leg Temperatures-Stable or Decreasing
- Core Exit TCs-Stable or Decreasing
- RCS Cold Leg Temperatures-At Saturation Temperature For SG Pressure

Per the question stem Core Exit Thermocouple indication is malfunctioning, which invalidates RCS Subcooling indications because it utilizes highest quadrant CETC in its calculation, and also the Core Exit TC direct indications are not valid. As a result, validation of natural circulation conditions must be based on SG Pressures-Stable or Decreasing, RCS Hot Leg Temperatures-Stable or Decreasing, and RCS Cold Leg Temperatures-At Saturation Temperature For SG Pressure

A is incorrect but plausible. The three parameters are correct, however SG Pressure should be "Stable or Decreasing" versus "Increasing".

B is incorrect but plausible. The three parameters are correct however RCS Hot Leg Temperatures should be "Stable or Decreasing" and RCS Cold Leg Temperatures should be "At Saturation Temperature For SG Pressure".

D is incorrect but plausible. All of the listed parameters in the distractor are ones that are indictors of natural circulation however the RCS Subcooling indication is not valid because CETC's are malfunctioning and are an input to the subcooling calculation.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ES-0.1, "Reactor Trip Response, Attachment I: Conditions that Indicate Natural Circulation Flow |

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L8058I05, L1225I08 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |

<table>
<tr><td>Question History:</td><td>Last NRC Exam</td><td colspan="3"></td></tr>
<tr><td rowspan="2">Question Cognitive Level:</td><td colspan="2">Memory or Fundamental Knowledge</td><td></td><td></td></tr>
<tr><td colspan="2">Comprehension or Analysis</td><td>X</td><td></td></tr>
<tr><td rowspan="2">10 CFR Part 55 Content:</td><td>55.41</td><td colspan="3">41.7</td></tr>
<tr><td>55.43</td><td colspan="3"></td></tr>
<tr><td colspan="5">Comments:</td></tr>
</table>

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#60** | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 028 Hydrogen Recombiner and Purge Control System (HRPS)<br>A1.01 Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Hydrogen concentration | |
| | Importance Rating | 3.4 | |

Proposed Question:

A large break LOCA has occurred. The crew is evaluating placing the Hydrogen Recombiners in service.

As containment hydrogen concentration rises, ___________ is the maximum hydrogen concentration per OS1023.40, "Hydrogen Recombiner Operation", allowing the Hydrogen Recombiners to be placed in operation.

A. 2.0%

B. 3.0%

C. 4.0%

D. 5.0%

Proposed Answer: C

Explanation (Optional):

C is correct. Per OS1023.40, "Hydrogen Recombiner Operation", "Containment air hydrogen concentration must be kept below 4 percent by volume to prevent the possibility of an explosion in containment."

A is incorrect but plausible. All distractors are numbers of reasonable value as compared to the correct answer, and are reasonable values of "lower explosive limits" for a multitude of various gasses."

B is incorrect but plausible. All distractors are numbers of reasonable value as compared to the correct answer, and are reasonable values of "lower explosive limits" for a multitude of various gasses.

D is incorrect but plausible. All distractors are numbers of reasonable value as compared to the correct answer, and are reasonable values of "lower explosive limits" for a multitude

| | | | | |
|---|---|---|---|---|
| of various gasses. | | | | |
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OS1023.40, Hydrogen Recombiner Operation | | | |
| Proposed references to be provided to applicants during examination: | | | | None |
| Learning Objective: | | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | **2015 Seabrook NRC Exam: One of previous two previous exams.** | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.10** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#61** | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 041 (SF4S SDS) Steam Dump/Turbine Bypass Control<br>041A4.08 - Ability to manually operate and/or monitor in the control room: Steam dump valves | |
| | Importance Rating | 3.0 | |

Proposed Question:

Given the following plant conditions:

- A plant cooldown is in progress.
- Reactor Coolant System loop Tavg:
  - Loop 1: 550°F and decreasing
  - Loop 2: 548°F and decreasing
  - Loop 3: 551°F and decreasing
  - Loop 4: 548°F and decreasing
- Steam header pressure is 1030 psig and decreasing.
- Steam Dump Mode Selector Switch is in Steam Pressure Mode.
- Steam Dump Controller is in Manual set at 30% demand.
- The operator momentarily places the Train 'A' and Train 'B' Steam Dump Bypass Interlock Switches to 'Bypass' and then releases them.

What is the status of the Steam Dump valves following the operator's actions?

A. All valves are fully closed.
B. Three valves in Group 1 are partially open.
C. Three valves in Group 1 are fully open and the valves in Group 2 are fully closed.
D. Three valves in Group 1 are fully open and three valves in Group 2 are partially open.

| Proposed Answer: | C |
|---|---|

Explanation (Optional):

C is correct. The P-12 Low-Low Setpoint is at 550°F. At this point taking the Steam Dumps to Bypass Interlock allows for operation of the Group 1 valves only. At 30% output on the

controller the Group 1 valves would be full open.

A is incorrect but plausible. The RCS loop temperatures listed in the stem of the question indicate that the P-12 setpoint/coincidence has been met. This would cause all the steam dump valves to close. If the student had a misconception regarding the operation of the Bypass Interlock switch operation they may believe that all of the valves would remain closed.

B is incorrect but plausible. It is true that the Group 1 valves would function however, with a 30% demand signal the Group 1 valves would be full open as then throttle through a 0-25% demand signal range.

D is incorrect but plausible. It is true that the Group 1 valves would be fully open however the Group 2 valves are not designed to function below P-12 even with Bypass Interlock actuated.

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|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • 1-NHY-509050, MS Dump Control w/ Functional Diagrams | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L8047I12, L8047I05, L8047I06 | | | |
| Question Source: | Bank # | X | Seabrook TEB 25201 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | 41.7 | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#62** | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 045 (SF 4S MTG) Main Turbine Generator<br>045A2.12 - Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Control rod insertion limits exceeded (stabilize secondary) | |
| | Importance Rating | 2.5 | |

Proposed Question:

Given the following conditions:

- The plant was initially at 100% power.
- A turbine runback occurred.
- The plant has stabilized at 82% power.
- D7762, "CTL ROD BANK D INSERTION LIMIT LO-LO" is in alarm.
- Control Bank D indicates 118 steps.

Which one of the following correctly completes the statement below?

The Technical Specification Limiting Condition for Operation (LCO) __(1)__ exceeded. The operator must immediately stop inserting rods and __(2)__.

| | (1) | (2) |
|---|---|---|
| A. | is | commence a rapid boration in accordance with OS1202.04, "Rapid Boration" |
| B. | is | commence a boration at greater than 50 GPM in accordance with OS1008.01, "Chemical and Volume Control Makeup Operations" |
| C. | is not | commence a rapid boration in accordance with OS1202.04, "Rapid Boration" |
| D. | Is not | commence a boration at greater than 50 GPM in accordance with OS1008.01, "Chemical and Volume Control Makeup Operations" |

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| Proposed Answer: | A | |

Explanation (Optional):

A is correct. Tech. Spec. 3.1.3.6, “Control Rod Insertion Limits” Limiting Condition for Operation states “The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT. The Rod Insertion Limit monitor and associated Lo-Lo alarms coincide with the COLR, Figure 1, “Control Bank Insertion Limits Versus Thermal Power”. The alarm response procedure for alarm D7762 prescribes rapid boration per OS1202.04.

B is incorrect but plausible. It is true that the Tech. Spec. LCO is exceeded. The 50GPM flow rate stated in the distractor is plausible as it is the required flow rate in response to rods not inserted post reactor trip.

C is incorrect but plausible. It is true that rabid boration should be commenced in accordance with OS1202.04, however the Tech. Spec. LCO is exceeded.

D is incorrect but plausible. The 50GPM flow rate stated in the distractor is plausible as it is the required flow rate in response to rods not inserted post reactor trip. Additionally, the Tech. Spec. LCO is exceeded.

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|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • Tech. Spec. 3.1.3.6, “Control Rod Insertion Limits”<br>• OS1202.04, “Rapid Boration”<br>• CORE OPERATING LIMITS REPORT, Figure 1, “Control Bank Insertion Limits Versus Thermal Power” |

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|---|---|
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L1197I10 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Turkey Point 2013 NRC Exam | | |

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |

| | | |
|---|---|---|
| 10 CFR Part 55 Content: | 55.41 | |
| | 55.43 | |

Comments:
This is a “bank” question that has been editorialized such that it applies to Seabrook Station.

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#63** | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | O55 Condenser Air Removal System (CARS)<br>K1.06 Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems: 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

Explanation (Optional):

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

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.

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

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|---|---|---|---|---|
| downpower however reactor trip and SI criteria are based on the threshold of maintaining >7% pressurizer level utilizing two charging pumps. | | | | |
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OS1227.02, Steam Generator Tube Leak | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L1190I02 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2009 NRC Remediation Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.2 to 41.9** | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#64** | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 056 (SF4S CDS) Condensate<br>056K1.03 - Knowledge of the physical connections and/or cause-effect relationships between the Condensate System and the following systems: 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 $\Delta P$ across the main feed reg. valves.

| Proposed Answer: | B |
|---|---|

Explanation (Optional):

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.

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.

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.

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|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • 1-NHY-503590<br>• 1-NHY-503591 | | | |
| Proposed references to be provided to applicants during examination: | | | | None |
| Learning Objective: | L8062I08 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2009 NRC Remedial Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | 41.2 to 41.9 | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#65** | Tier # | 2 | |
| | Group # | 2 | |
| | K/A # | 068 (SF9 LRS) Liquid Radwaste<br>068K4.01 - Knowledge of design feature(s) and/or interlock(s) which provide for the following: Safety and environmental precautions for handling hot, acidic and radioactive liquids | |
| | Importance Rating | 3.4 | |

Proposed Question:

Given the following plant conditions:

- A release of the 'B' Waste Test Tank is in progress at 45 gallons per minute to the Discharge Transition Structure.
- RDMS Channel R6509(1LM621), Liq Waste Tk to CW Sys fails to the HIGH ALARM condition.

What action will occur?

A. ONLY 1-WL-FCV-1458-1, High Capacity Waste Distillate to Discharge Transition Structure Valve closes.

B. ONLY 1-WL-FCV-1458-2, Low Capacity Waste Distillate to Discharge Transition Structure Valve closes.

C. 1-WL-FCV-1458-1, High Capacity Waste Distillate to Discharge Transition Structure Valve closes and the 'B' Waste Test Tank Pump trips.

D. 1-WL-FCV-1458-2, Low Capacity Waste Distillate to Discharge Transition Structure Valve closes and the 'B' Waste Test Tank Pump trips.

Proposed Answer: A

Explanation (Optional):

A is correct. A High Radiation signal from RM-6509 will close the modulating Waste Test Tank Discharge Valve. The WTT discharge path has a low capacity valve, 1-WL-FCV-1458-2, which is utilized for discharges where the flow rate is </= 20 gpm. The high capacity valve, 1-WL-FCV-1458-1 is utilized for flow rates > 20 gpm. The stem of the question states that the discharge flow rate is 45 gpm so 1-WL-FCV-1458-1 would be the valve being utilized.

The automatic closure of the discharge valve provides a precaution to ensure that Offsite

Dose Calculation Manual, part 6.0: Radioactive Liquid Effluents, item 6.1: Concentration is adhered to.

B is incorrect but plausible. A High Radiation signal from RM-6509 will close the modulating Waste Test Tank Discharge Valve and would close 1-WL-FCV-1458-2 if it were in service however the stem of the question states that the flow rate is 45 gpm. In this case 1-WL-FCV-1458-1 would be the in service valve.

C is incorrect but plausible. It is true that 1-WL-FCV-1458-1 would close however the high radiation signal does not cause the Waste Test Tank pumps to trip.

D is incorrect but plausible. A High Radiation signal from RM-6509 will close the modulating Waste Test Tank Discharge Valve and would close 1-WL-FCV-1458-2 if it were in service however the stem of the question states that the flow rate is 45 gpm. In this case 1-WL-FCV-1458-1 would be the in service valve. Additionally, the high radiation signal does not cause the Waste Test Tank pumps to trip.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • 1-NHY-504062, WL-Liquid Effluent Discharge Logic Diagram<br>• 1-NHY-504060, WL-Waste Test Tank Pumps Logic Diagram |
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | **2013 Seabrook NRC Exam: One of last two previous exams.** | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |

| | | |
|---|---|---|
| 10 CFR Part 55 Content: | 55.41 | **41.7** |
| | 55.43 | |
| Comments: | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#66** | Tier # | 3 | |
| | Group # | 1 | |
| | K/A # | G2.1.37<br>Knowledge of procedures, guidelines or limitations associated with reactivity management | |
| | Importance Rating | 4.3 | |

Proposed Question:

Per procedure OS1000.07, "Approach to Criticality" what reactivity control parameters are verified prior to each 50 step incremental control rod withdrawal?

A. RCS boron is verified to be within 15 ppm of critical boron. All Shutdown Rods are verified to be fully withdrawn.
B. RCS boron is verified to be within 15 ppm of critical boron. Critical Rod Height is predicted to be above the Rod Insertion Limit.
C. All Shutdown Rods are verified to be fully withdrawn. The lowest operating loop $T_{avg}$ is verified to be greater than 551°F.
D. The lowest operating loop $T_{avg}$ is verified to be greater than 551°F. Critical Rod Height is predicted to be above the Rod Insertion Limit.

| Proposed Answer: | C |
|---|---|

Explanation (Optional):

C is correct. Per OS1000.07, Approach to Criticality, step 4.4.5, all shutdown rods are verified withdrawn within 15 minutes of any control bank withdrawal. Additionally, the lowest operating loop $T_{avg}$ is verified to be greater than 551°F every 15 minutes until the reactor is declared critical. OS1000.07 includes specific steps to perform these verifications prior to each 50 step incremental control rod withdrawal. These verifications are pursuant to technical specification reactivity limitations associated with a) ensuring adequate Shutdown Margin and b) ensuring that moderator temperature coefficient is within its analyzed temperature range.

A is incorrect but plausible. It is true that all shutdown rods are verified withdrawn within 15 minutes of any control bank withdrawal. It is also true that procedure OS1000.07 directs verifying RCS boron concentration within 15 ppm of critical boron, however this is required to be done within 4 hours of performing an approach to criticality as part of verifying that the Estimated Critical Position data is accurate.

B is incorrect but plausible. It is true that procedure OS1000.07 directs verifying RCS boron concentration within 15 ppm of critical boron, however this is required to be done within 4 hours of performing an approach to criticality as part of verifying that the Estimated Critical Position data is accurate. It is also true that procedure OS1000.07 directs verifying that the estimated critical rod height is predicted to be above the rod insertion limit, however this is required to be done within 4 hours of performing an approach to criticality as part of verifying that the Estimated Critical Position data is accurate.

D is incorrect but plausible. It is true that the lowest operating loop $T_{avg}$ is verified to be greater than 551°F prior to each incremental control rod withdrawal. It is also true that procedure OS1000.07 directs verifying that the estimated critical rod height is predicted to be above the rod insertion limit, however this is required to be done within 4 hours of performing an approach to criticality as part of verifying that the Estimated Critical Position data is accurate.

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|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OS1000.07, Approach to Criticality | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2010 NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | 41.1 | | |
| | 55.43 | 43.6 | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#67** | Tier # | 3 | |
| | Group # | 1 | |
| | K/A # | G2.1.42<br>Knowledge of new and spent fuel movement procedures | |
| | Importance Rating | 2.5 | |

Proposed Question:

Given the following conditions:

- Refueling operations are being performed per OS1000.09, "Refueling Operation".
- A Control Room Operator notices that the Source Range Audio Count Rate Monitor in the control room is no longer functioning.

What actions should be taken?

A. Refueling operations may continue using Gammametrics SR indications.

B. Immediately suspend all operations involving core alterations or positive reactivity changes.

C. Refueling operations may continue if an operable audio count rate monitor is available in containment.

D. Ensure the boron concentration in the RCS is greater than the Refueling Boron Concentration Limits of the COLR once per 12 hours and suspend any activities which could dilute the RCS.

Proposed Answer: B

Explanation (Optional):

B is correct. As described in OS1000.09, Section 2.2 of the procedure prerequisites, one source range NI instrument must have audible indication in containment and in the Control Room. Per T.S.3.9.2 continuous audible indication is required in the containment and the control room. If not immediately suspend core alterations or positive reactivity changes.

A is incorrect but it is plausible that using alternate indications of SR refueling may

continue.

C is incorrect but it is plausible that with audible count rate indicating in the containment then the SR is operable and refueling operations could continue.

D is incorrect but plausible as this is part of the required actions of T.S.3.9.2 action b if both SR instruments are inoperable.

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|---|---|---|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1000.09, "Refueling Operation"<br>• T.S.3.9.2 Refueling Operations Instrumentation | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L8031I12 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | **2013 Seabrook NRC Exam: One of last two previous exams.** | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | | | |
| | 55.43 | | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#68** | Tier # | 3 | |
| | Group # | 2 | |
| | K/A # | G2.2.15<br>Ability to determine the expected plant configuration using design and configuration control documentation | |
| | Importance Rating | 3.9 | |

Proposed Question:

You are performing an independent verification (IV) on a normally LOCKED OPEN manual valve. What is the proper method of verifying the valve position?

A. Observe the valve stem position and verify that the locking device is properly installed.

B. Check the valve operator indicates in the OPEN position and verify that the locking device is properly installed.

C. Remove the locking device, check the operator in the CLOSED direction, return the valve to full open, and reinstall the locking device.

D. Verify that the locking device is properly installed AND while leaving the locking device installed check the operator in the CLOSED direction.

Proposed Answer: D

Explanation (Optional):

D is correct per OS1090.05, “Component Configuration Control”, Section 4.4.3, Position Verification of Manual Valves.

A is incorrect but plausible. It is true that valve stem position is an indicator of valve position, however this method is not prescribed in OS1090.05, Section 4.4.3, Position Verification of Manual Valves.

B is incorrect but plausible. It is true that some valve operators provide an indication of valve position, however this method is not prescribed in OS1090.05, Section 4.4.3, Position Verification of Manual Valves.

C is incorrect but plausible. It is true that you would check the operator in the CLOSED direction, however OS1090.05, Section 4.4.3, Position Verification of Manual Valves does not prescribe removing the locking device.

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|---|---|---|---|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1090.05, "Component Configuration Control" | | | | |
| Proposed references to be provided to applicants during examination: | | | | None | |
| Learning Objective: | | | | | |
| Question Source: | Bank # | X | Seabrook TEB #16258 | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | | | | |
| Question History: | Last NRC Exam | | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | X | |
| | Comprehension or Analysis | | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.10** | | | |
| | 55.43 | **43.3** | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#69** | Tier # | 3 | |
| | Group # | 2 | |
| | K/A # | G2.2.37<br>Ability to determine operability and/or availability of safety related equipment | |
| | Importance Rating | 3.6 | |

Proposed Question:

Given the following conditions:

- RCS leakage into Steam Generator "A" is .05 gallons per minute.
- RCS identified leakage is 1.2 gallons per minute
- RCS unidentified leakage is 1.8 gallons per minute.

Which RCS Leakage Limiting Condition for Operation has been exceeded, if any?

A. None
B. RCS identified leakage
C. RCS unidentified leakage
D. Steam Generator tube leakage

| Proposed Answer: | C |
|---|---|

Explanation (Optional):

C is correct. Per Tech. Spec. 3.4.6.2 RCS unidentified leakage must be less than 1 gpm.

All of the distractors are plausible as they are modes of leakage identified in Tech. Spec. 3.4.6.2, however all of the distractor leak rates satisfy Limiting Condition for Operation requirements.

| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • Tech Spec 3.4.6.2, "Operational Leakage" |
|---|---|
| Proposed references to be provided to applicants during examination: | None |
| Learning | L8010I07 |

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|---|---|---|---|---|
| Objective: | | | | |
| Question Source: | Bank # | X | Seabrook 2003 Company Exam | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.7** | | |
| | 55.43 | **43.5** | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#70** | Tier # | 3 | |
| | Group # | 3 | |
| | K/A # | G2.3.13<br>Knowledge of radiological safety procedures pertaining to licensed operator duties | |
| | Importance Rating | 3.4 | |

Proposed Question:

An accessible area of the Primary Auxiliary Building has a general area dose rate of 500 mrem/hr. What posting should be displayed at the entrance to this area?

A. Radiation Area
B. High Radiation Area
C. Very High Radiation Area
D. Tech. Spec. Locked High Radiation Area

Proposed Answer: B

Explanation (Optional):

B is correct per SSRP, Chapter 1, Section 3.4.6, Area Designations.

A,C, and D are all incorrect but plausible. The following are the criteria for area designations:

Radiation Area-Any area accessible to individuals in which radiation levels could result in the individual receiving a dose in excess of 5mrem (DDE) in one-hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

High Radiation Area- Any area accessible to individuals in which radiation levels could result in the individual receiving a dose in excess of 100mrem (DDE) in one-hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

- Very High Radiation Area-A high radiation area accessible to individuals, in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads in one-hour at 1 meter from the radiation source or from any surface

that the radiation penetrates.

Tech. Spec. Locked High Radiation Area- Any high radiation area (1) accessible to individuals in which radiation levels could result in an individual receiving a dose equivalent of >1000mrem (DDE) in one-hour at 30 cm from the radiation source or from any surface that radiation penetrates and (2) not meeting the requirements of a Very High Radiation Area.

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • SSRP, Chapter 1, Section 3.4.6, Area Designations |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L1525I09 |

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|---|---|---|---|---|
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2010 NRC Exam | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | 41.12 | |
| | 55.43 | 43.3 | |
| Comments: | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#71** | Tier # | 3 | |
| | Group # | 3 | |
| | K/A # | G2.3.14<br>Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities | |
| | Importance Rating | 3.4 | |

| Proposed Question: | |
|---|---|

Given the following plant conditions:

- 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 | |
|---|---|---|

Explanation (Optional):

C is correct per the Westinghouse basis document for E-3.

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.

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.

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| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • Westinghouse Background Document, E-3. | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L1205I03 | | | |
| Question Source: | Bank # | X | Seabrook 2009 NRC Remedial Exam | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.12** | | |
| | 55.43 | **43.4** | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#72** | Tier # | 3 | |
| | Group # | 3 | |
| | K/A # | G2.3.4<br>Knowledge of radiation exposure limits under normal and emergency conditions | |
| | Importance Rating | 3.2 | |

Proposed Question:

Given the following conditions:

- An RCS leak has occurred inside of Containment requiring a plant shutdown.
- A Planned Special Exposure is being created to allow a NSO to isolate the leak.
- The NSO's current Total Effective Dose Equivalent (TEDE) dose is:
    - Exposure for the year: 100 mrem
    - Current total lifetime exposure: 1200 mrem

What is the maximum exposure that the NSO may receive (mrem TEDE)?

A. 800

B. 1900

C. 5000

D. 13800

Proposed Answer: C

Explanation (Optional):

C is correct. Per RP5.2, Figure 5.2, Federal Planned Special Exposure Annual and Lifetime Limits, a planned special exposure can result in a maximum one year dose of 5 R TEDE or a maximum lifetime dose of 25R TEDE

A is incorrect but plausible. The normal administrative limit for annual exposure is 2000 mrem TEDE. The student may choose this distractor if they believe that the maximum dosage is the admin annual limit minus the NSOs current lifetime exposure.

B is incorrect but plausible. The normal administrative limit for annual exposure is 2000 mrem TEDE. The student may choose this distractor if they believe that the maximum dosage is the admin annual limit minus the NSOs annual exposure.

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| D is incorrect but plausible. The PSE limit for eye lens is 15R. The student may choose this distractor if they believe that the maximum dosage 15R minus the NSOs lifetime exposure. | | | | |
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • RP5.2, Figure 5.2, Federal Planned Special Exposure Annual and Lifetime Limits | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L1307I06 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | X | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | **41.12** | | |
| | 55.43 | **43.3** | | |

<u>Parent Question:</u>

The following plant conditions exist:

- 100% power.
- Blackout conditions are in effect throughout the eastern U.S.
- Seabrook Station remains online.
- RCS leak has occurred inside of Containment , requiring a plant shutdown.
- A Planned Special Exposure has been created to allow the NSO to isolate the RCS leak, allowing the plant to stay online.
- The operators current TEDE dosage is:
    - Exposure for the year: 200 mrem
    - Current total lifetime exposure: 1200 mrem

What is the maximum exposure this operator may receive?

A. 4800 mrem

B. 5000 mrem

C. 10000 mrem

D. 25000 mrem

Answer: B

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#73** | Tier # | 3 | |
| | Group # | 4 | |
| | K/A # | G2.4.12<br>Knowledge of general operating crew responsibilities during emergency operations. | |
| | Importance Rating | 4.0 | |

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 and 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

Explanation (Optional):

B is correct. Per OP-9.2, Transient Response Procedure Users Guide, EOP rules of usage dictate immediately transitioning to the Highest RED path condition. The Core Cooling Critical Safety

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| Function is a higher priority function than the Containment Critical Safety Function.<br><br>All of the distractors are plausible if the student misapplies the rules of usage contained in OP-9.2. |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OP-9.2, Transient Response Procedure Users Guide |
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L1195I05 | | | |
| Question Source: | Bank # | X | Seabrook TEB# 32278 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 Content: | 55.41 | **41.10** | |
| | 55.43 | | |
| Comments: | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#74** | Tier # | 3 | |
| | Group # | 4 | |
| | K/A # | G2.4.5<br>Knowledge of the organization of the operating procedures network for normal, abnormal and emergency evolutions. | |
| | Importance Rating | 3.7 | |

Proposed Question:

Given the following:

- The reactor trips from 100% power.
- Concurrent with the reactor trip controlling PZR level instrument L-459 fails low and letdown isolates.

How is the crew required to implement the EOPs/AOPs?

A. The crew enters E-0, "Reactor Trip or Safety Injection". After completing step 4, the crew transitions to ES-0.1, "Reactor Trip Response" and completes OS1201.07, "PZR Level Instrument Failure" in parallel with ES-0.1.

B. The US hands the WCS OS1201.07, "PZR Level Instrument Failure", who directs the BOP to perform the steps of the AOP in parallel with the US entering E-0, "Reactor Trip or Safety Injection" and directs the PSO to perform the steps of E-0.

C. The crew enters E-0, "Reactor Trip or Safety Injection". After completing step 4, the crew transitions to OS1201.07, "PZR Level Instrument Failure", when AOP is completed the crew transitions to ES-0.1, "Reactor Trip Response".

D. The crew enters E-0, Reactor Trip or Safety Injection. After processing through and exiting emergency procedures, abnormal procedures can be entered and OS1201.07, PZR Level Instrument Failure will be performed.

| Proposed Answer: | A |
|---|---|

Explanation (Optional):

A is correct. OP 9.2 section 4.9.4 states there are times when the use of other procedures will provide written guidance to respond to a condition not addressed in the EOP network. Parallel use of abnormal procedures is allowed following completion of immediate actions.

B is incorrect. It is plausible that as immediate action steps are being completed by the

PSO and the US an additional crew member and the BOP could perform abnormal procedure actions.

C is incorrect. It is plausible to transition to the abnormal procedure as this is done if not in the Emergency procedures.

D is incorrect but plausible that emergency procedures take priority and abnormal procedures cannot be processed until emergency procedures are complete.

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|---|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OP 9.2, Transient Response Procedure Guide | | | | |
| Proposed references to be provided to applicants during examination: | | | | None | |
| Learning Objective: | L1505I10 | | | | |
| Question Source: | Bank # | X | | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | | | | |
| Question History: | Last NRC Exam | **2013 Seabrook NRC Exam: One of previous two previous exams.** | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | X | |
| | Comprehension or Analysis | | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.10** | | | |
| | 55.43 | | | | |
| Comments: | | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#75** | Tier # | 3 | |
| | Group # | 4 | |
| | K/A # | G2.4.6<br>Knowledge of symptom based 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

Explanation (Optional):

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

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| should not be established until consulting with the TSC.<br><br>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.<br><br>D is incorrect but plausible. This answer states the technique used if RCS temperature is stable or decreasing. |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • FR-H.1, Response to Loss of Secondary Heat Sink |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L1211I02 |

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|---|---|---|---|---|
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2009 NRC Remedial Exam | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 Content: | 55.41 | **41.10** | |
| | 55.43 | | |
| Comments: | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#76** | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 000015 (APE 15) Reactor Coolant Pump Malfunctions / 4<br>015AG2.1.20 - Ability to execute procedure steps. | |
| | Importance Rating | | 4.6 |

Proposed Question:

Given the following plant conditions:

- The plant is at 45% power.
- VAS alarm D5782, "RCP D Motor Frame Vibration High" is in alarm.
- Reactor Coolant Pump 'D' frame vibration is 4 mils and increasing at 1.0 mil per hour.
- Reactor Coolant Pump 'D' shaft vibration is 10 mils and increasing at 1.0 mil per hour.
- Vibration values have been determined to be valid.

What action is required?

A. Shutdown the plant to Mode 3 and then stop the 'D' RCP per OS1001.05, "Reactor Coolant Pump Operations."

B. Continue to monitor 'D' RCP vibration and notify Tech. Support per D5782, "RCP D Motor Frame Vibration High" alarm response procedure.

C. Feed the 'D' Steam Generator to between 60 and 70% NR level and trip the 'D' RCP in accordance with OS1201.01, "RCP Malfunction."

D. Trip the reactor and enter E-0 "Reactor Trip or Safety Injection." Stop the 'D' RCP after the immediate action steps are complete.

| Proposed Answer: | C |
|---|---|

Explanation (Optional):

C is correct. The 'D' RCP frame vibration is above the alert value of "3 mils and increasing at greater then 0.2 mils per hour". Plant power is below the P-8 permissive value (50% power-reset @ 48%). In this case OS1201.01 directs feeding up the steam generators and removing the pump from service.

All of the distractors are plausible as they require application of procedure requirement knowledge and the understanding of the P-8 permissive relay.

A is incorrect. It is correct that the procedure directs a plant shutdown to Mode 3 however the RCP is removed from service first.
B is incorrect. If the frame vibration level was below the alert value then the abnormal procedure would direct continued monitoring of RCP parameters and notification of Tech. Support.

D is incorrect. If plant power were above P-8 (52%) then the procedure would direct tripping the reactor.

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|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OS1201.01, "RCP Malfunction" | | | |
| Proposed references to be provided to applicants during examination: | | | | None |
| Learning Objective: | L1181I03 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2010 NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | 41.10 | | |

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|---|---|---|---|
| | 55.43 | 43.5 | |

Comments:



Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)? — Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions? — Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs? — Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? — Yes → RO question

No

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#77** | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 000054 (APE 54; CE E06) Loss of Main Feedwater /4<br>054AG2.4.35 - Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects | |
| | Importance Rating | | 4.0 |

Proposed Question:

Given the following conditions:

- The plant was initially at 100% power with all plant systems in normal lineup.
- A loss of feedwater event occurred coincident with a loss of Bus 5.
- The crew is implementing FR-H.1, "Response to Loss of Secondary Heat Sink.
- Reactor Coolant System Pressure is 2350 psig and rising.
- Steam Generator pressures are all at 1125 psig
- Steam Generator Wide Range Levels are:
  - SG "A": 11% and lowering
  - SG "B": 11% and lowering
  - SG "C": 35% and slowly lowering
  - SG "D": 35% and slowly lowering
- Condensate Storage Tank level is 280,000 gallons.
- All Reactor Coolant Pumps are running.
- The electric driven EFW pump has a sheared shaft.
- The turbine driven EFW pump has tripped on overspeed.
- The Roving NSO has notified the crew that the Steam Driven EFW Pump trip valve is mechanically bound and assistance from a maintenance mechanic is needed.

Per FR-H.1, which of the following is the correct procedural strategy to establish feed to the steam generators?

A. Locally restore EFW flow per OS1036.03, "Resetting the Steam Driven EFW Pump Trip Valve". When the Steam Driven EFW Pump has been restarted then commence feeding all of the Steam Generators.

B. Start the Startup Feedwater Pump. Direct an NSO to open CO-V-142, "Condensate

Storage Tank Lower Suction Tap. Commence feeding all of the Steam Generators with the Startup Feedwater Pump.

C. Stop all Reactor Coolant Pumps. Direct an NSO to swap the Startup Feedwater Pump breaker to Bus 6. Commence feeding the "C", or "D" Steam Generators with the Startup Feedwater Pump.

D. Stop all Reactor Coolant Pumps. Establish feed flow from the condensate system by depressurizing the RCS, depressurizing at least one Steam Generator, locally bypassing the Main Feed Pump and SUFP and establish flow to the depressurized steam generator.

| | | |
|---|---|---|
| Proposed Answer: | D | |

Explanation (Optional):

D is correct. Per FR-H.1 the correct action to take is to establish feed flow from the condensate pumps as the other options are not readily available.

All of the distractors are plausible as they are options contained in FR-H.1, however distractor "A" is not correct because the crew should not wait to get the steam driven EFW and distractors B and C are not correct as the SUFP is not readily available.

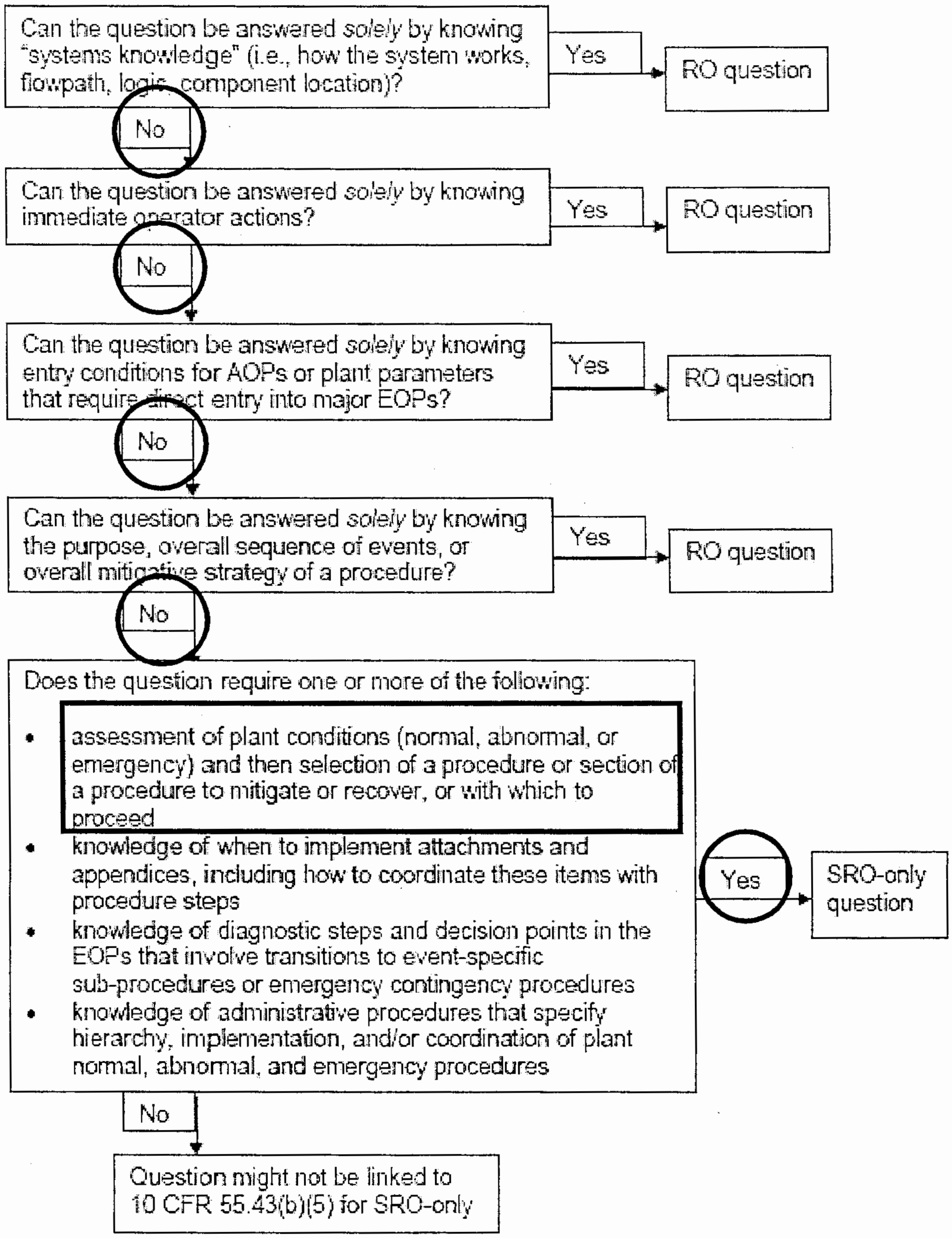
| | |
|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • FR-H.1, "Response to Loss of Secondary Heat Sink" |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L1211I02, L1211I03 |

| | | | | |
|---|---|---|---|---|
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |

| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 | 55.41 | **41.10** | |

| | | |
|---|---|---|
| | 55.43 | **43.5** |
| Comments: | | |

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#78** | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 000055 (EPE 55) Station Blackout / 6<br>055EA2.04 - Ability to determine or interpret the following as they apply to a Station Blackout: Instruments and controls operable with only dc battery power available | |
| | Importance Rating | | 4.1 |

Proposed Question:

Given the following conditions:

- The plant was initially at 100% power.
- A station blackout (Loss of All AC) has occurred due to an earthquake.
- The "A" Emergency Diesel Generator is out of service for maintenance and is expected to be available in 9 hours.
- The "B" Emergency Diesel Generator tripped due to a broken crank shaft and the NSO has reported that there is a large quantity of oil on the floor.
- The electrical dispatcher has informed the crew that offsite AC power will be available in 6 hours.
- The crew is implementing ECA-0.0, "Loss of All AC Power" and has re-energized Bus 6 from SEPS per procedure step 5b.
- The crew is now implementing procedure Step 5i, "Check AC emergency bus-ENERGIZED BY EMERGENCY DIESEL GENERATOR".

What indication was the crew able to use to verify that the reactor was tripped and per ECA-0.0, Step 5i, what procedural action should the crew take next?

A. The crew was able to verify that the reactor was tripped using main control board Reactor Trip breaker indications. The crew should declare ELAP (Extended Loss of AC Power) and go to FSG-0.0, Extended Loss of All AC Power With SEPS".

B. The crew was able to verify that the reactor was tripped using main control board Reactor Trip breaker indications. The crew can now return to the procedure and step in effect and should implement Functional Restoration Procedures as required.

C. The crew was able to verify that the reactor was tripped using Digital Rod Position Indication (DRPI) rod bottom light indications. The crew should declare ELAP (Extended Loss of AC Power) and go to FSG-0.0, Extended Loss of All AC Power With SEPS".

D. The crew was able to verify that the reactor was tripped using Digital Rod Position

| |
|---|
| Indication (DRPI) rod bottom light indications. The crew can now return to the procedure and step in effect and shall implement Functional Restoration Procedures as required. |

| Proposed Answer: | A | |
|---|---|---|
| Explanation (Optional): | | |

A is correct. Reactor Trip switch light indications are still available as they are powered from DC power panels PP-111A and 111B. Additionally, ECA-0.0, Step 5i states "If at least one AC emergency bus can NOT be energized from an emergency diesel generator or offsite power source within 4 hours of the SBO event" then declare ELAP (Extended Loss of AC Power) and go to FSG-0.0, Extended Loss of All AC Power With SEPS".

B is incorrect but plausible. It is true that Reactor Trip switch light indications are still available as they are powered from DC power panels PP-111A and 111B. If an AC emergency bus can be energized by an emergency diesel generator then the crew could return to procedure and step in affect, however Bus 6 is powered from SEPS vice an EDG.

C is incorrect but plausible. It is true that the crew should declare ELAP (Extended Loss of AC Power) and go to FSG-0.0, Extended Loss of All AC Power With SEPS". Additionally, although Digital Rod Position Indication is one of the procedurally prescribed indications to utilize to verify a reactor trip, this source of indication is powered from AC MCC 531 which is powered from Bus 5.

D is incorrect but plausible. If an AC emergency bus can be energized by an emergency diesel generator then the crew could return to procedure and step in affect, however Bus 6 is powered from SEPS vice an EDG. Additionally, although Digital Rod Position Indication is one of the procedurally prescribed indications to utilize to verify a reactor trip, this source of indication is powered from AC MCC 531 which is powered from Bus 5.

| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • ECA-0.0, "Loss of All AC Power"<br>• 1-NHY-310944, Sheet HD2a, Reactor Trip Breaker.<br>• 1-NHY-310944, Sheet HD3a, Reactor Trip Breaker. |
|---|---|

| Proposed references to be provided to applicants during examination: | None |
|---|---|

| Learning Objective: | L8056I10, L8056I11, L8032I06, L8067I03 |
|---|---|

| | | | | |
|---|---|---|---|---|
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |

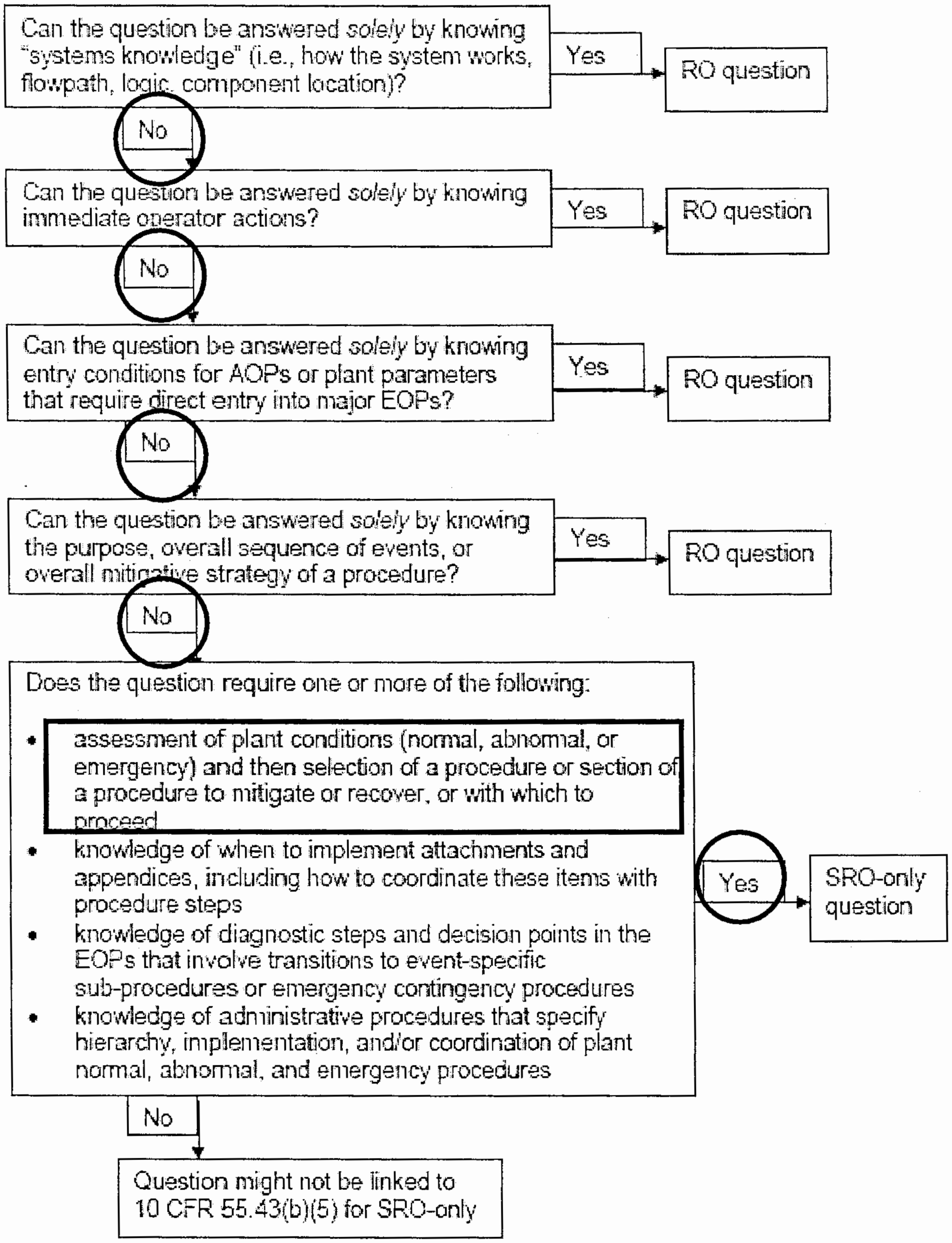
| | | | |
|---|---|---|---|
| Question History: | Last NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 | 55.41 | | |

| | | |
|---|---|---|
| | 55.43 | **43.5** |

Comments:

ES-401 8 Attachment 2

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#79** | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 000058 (APE 58) Loss of DC Power / 6<br>058AG2.4.20 - Knowledge of operational implications of EOP warnings, cautions and notes. | |
| | Importance Rating | | 4.3 |

Proposed Question:

While performing ECA-0.0, "Loss of All AC Power", the TSC directs the operators to shed non-vital DC loads per Attachment H . What is the reason for performing that action?

A. Prevent DC fuse failure or breaker trips due to high current flow.

B. Lower battery hydrogen generation rates while no ventilation is available.

C. Lower the rate of battery discharge to conserve vital battery power supply.

D. Ensure that DC oil pumps are still running until the main turbine has stopped rolling.

Proposed Answer: C

Explanation (Optional):

C is correct per Westinghouse background document "ECA-0.0, Background", page 112.

A is incorrect but plausible. Plant fuses and breakers are designed to handle full bus load.

B is incorrect but plausible. It is plausible for the batteries to generate hydrogen however this is not the reason for the action.

D is incorrect but plausible. The lube oil pumps are powered from non-vital DC, however this is not the reason for the action.

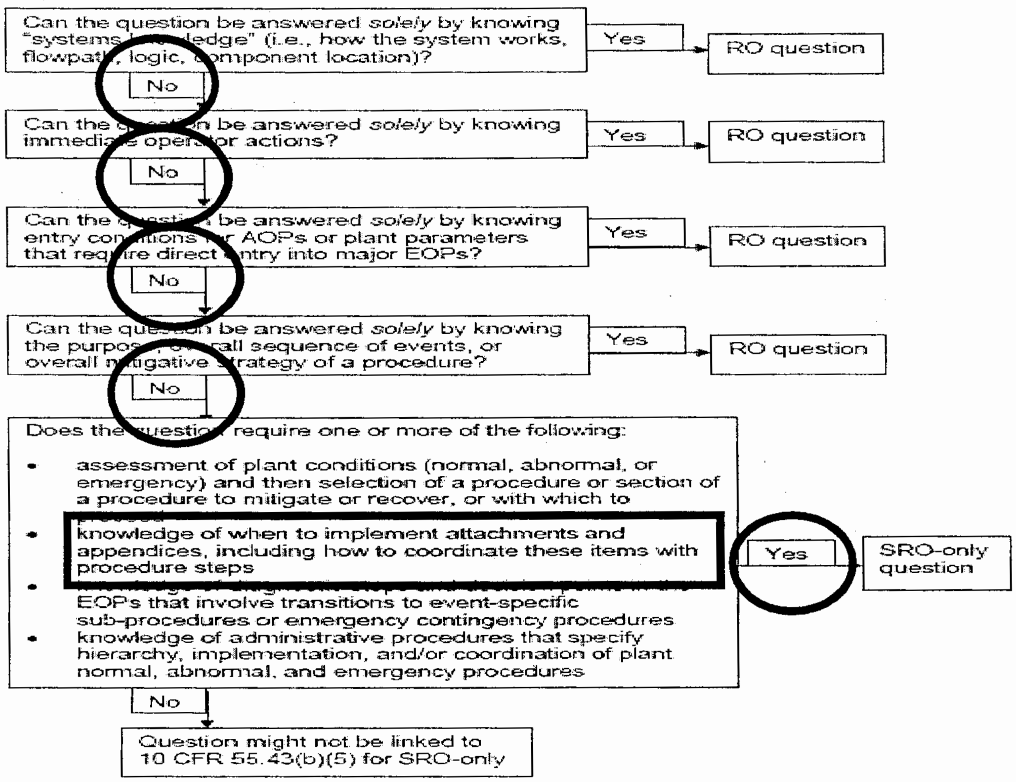
| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ECA-0.0, "Loss of All AC Power"<br>• Westinghouse background document "ECA-0.0, Background", page 112 |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L8607I03 |
| Question Source: | Bank # |

| | | | | |
|---|---|---|---|---|
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | **41.10** | | |
| | 55.43 | **43.5** | | |

Comments:



Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#80** | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | 000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6<br>077AA2.10 - Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Generator overheating and required actions | |
| | Importance Rating | | 3.8 |

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- An electrical grid disturbance has resulted in the following conditions:
  - 345kV system voltage is 343.5 kV and slowly lowering.
  - Emergency Bus 5 voltage is 3550 Volts and slowly lowering.
  - Emergency Bus 6 voltage is 3590 Volts and slowly lowering.

Which of the following correctly completes the statement below?

The main generator winding temperatures will (1) and OS1246.02, "Degraded Vital AC Power (Plant Operating), Step 16, "Check For Sustained Degraded Bus Voltage Condition" will require the crew to (2).

| | (1) | (2) |
|---|---|---|
| A. | Rise | Transfer ONLY Emergency Bus 5 power source to DG 1A per OS1246.02, Attachment R, "Removing Emergency Bus 5 From Offsite Power Source" |
| B. | Lower | Transfer ONLY Emergency Bus 5 power source to DG 1A per OS1246.02, Attachment R, "Removing Emergency Bus 5 From Offsite Power Source" |
| C. | Rise | Trip the Reactor. Go to E-0, "Reactor Trip or Safety Injection". When E-0 immediate actions are complete transfer emergency busses 5 AND 6 to the Emergency Diesel Generators. |

| | | |
|---|---|---|
| D. | Lower | Trip the Reactor. Go to E-0, "Reactor Trip or Safety Injection". When E-0 immediate actions are complete transfer emergency busses 5 AND 6 to the Emergency Diesel Generators. |

| | | |
|---|---|---|
| Proposed Answer: | C | |

Explanation (Optional):

C is correct. If grid voltage drops main generator winding current will increase resulting in rising temperature. Additionally, per OS1246.02, "Degraded Vital AC Power (Plant Operating), Step 16, if both Emergency Bus voltages are less than 3600 volts then the procedure directs the crew to Trip the Reactor, Go to E-0, "Reactor Trip or Safety Injection", and when E-0 immediate actions are complete transfer emergency busses 5 AND 6 to the Emergency Diesel Generators.

A is incorrect but plausible. It is true that if grid voltage drops main generator winding current will increase resulting in rising temperature. Additionally, OS1246.02 does provide direction to transfer a single emergency bus to its Emergency Diesel Generator, however that is for the case where only one of the emergency busses is impacted by the degraded grid voltage. Given the conditions in the question stem, both emergency busses are impacted.

B is incorrect but plausible. It is true that OS1246.02 does provide direction to transfer a single emergency bus to its Emergency Diesel Generator, however that is for the case where only one of the emergency busses is impacted by the degraded grid voltage. Given the conditions in the question stem, both emergency busses are impacted. It is not true that main generator temperature would lower.

D is incorrect but plausible. It is true that per OS1246.02, "Degraded Vital AC Power (Plant Operating), Step 16, if both Emergency Bus voltages are less than 3600 volts then the procedure directs the crew to Trip the Reactor, Go to E-0, "Reactor Trip or Safety Injection", and when E-0 immediate actions are complete transfer emergency busses 5 AND 6 to the Emergency Diesel Generators. It is not true that main generator temperature would lower.

| | |
|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OS1246.02, "Degraded Vital AC Power (Plant Operating) |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L1199I05, L1199I10, L1199I11 |

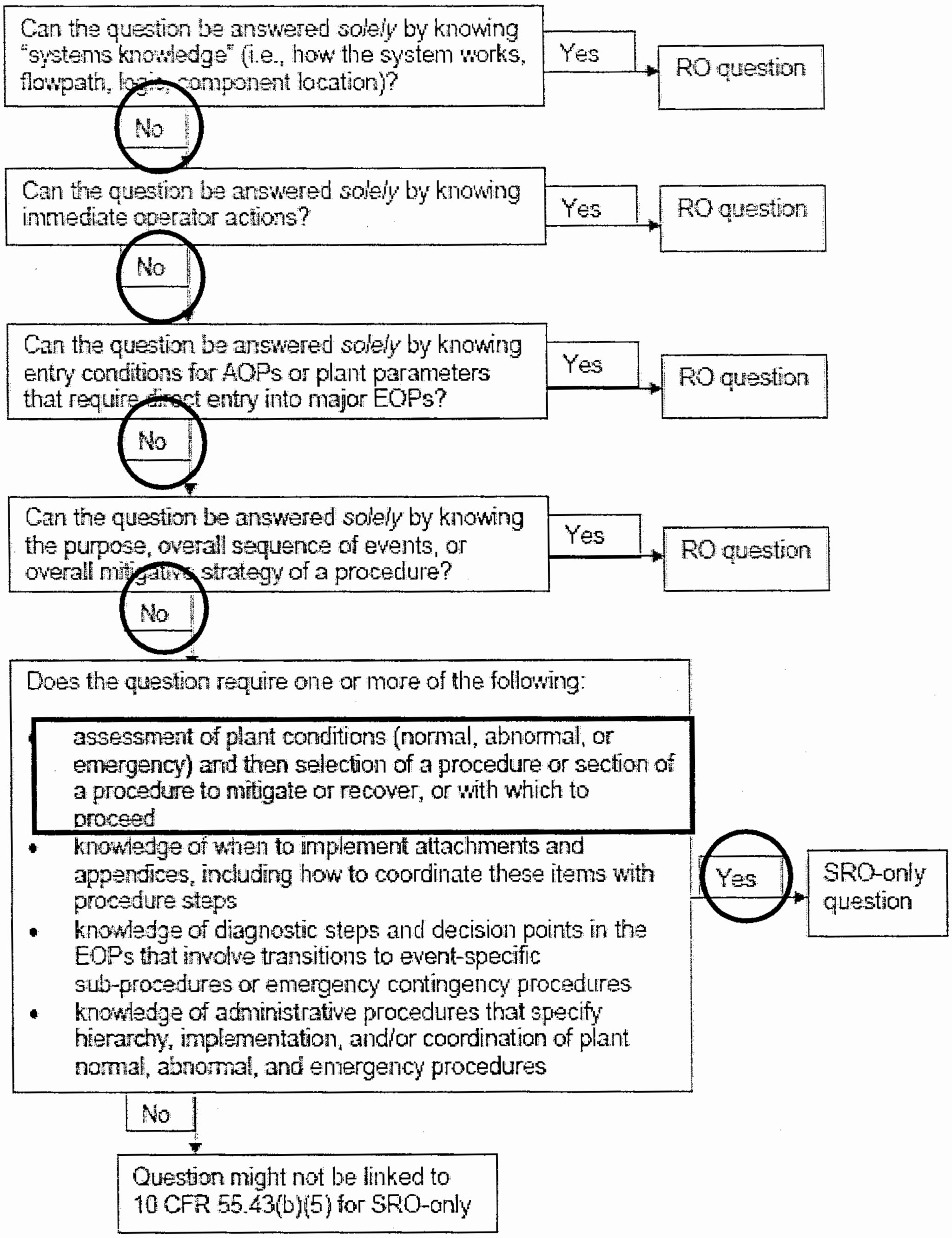
| | | | | |
|---|---|---|---|---|
| Question Source: | Bank # | | | |
| | Modified Bank# | X | | (Note changes or attach |

| | | | | |
|---|---|---|---|---|
| | | | | Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Farley 2011 NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | 41.5 | | |

| | | |
|---|---|---|
| | 55.43 | **43.5** |



Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



Parent Question:

Unit 1 is at 100% power.

Several substations are separated from the grid resulting in the following plant conditions:

- Unit 1 Generator Voltage is 20.45kV
- The following alarms have actuated:
  - WE2, 1 F 4KV Bus OV-OR-UV or Loss of DC.
  - VE2, 1 G 4KV Bus OV-OR-UV or Loss of DC.
- Grid frequency has fallen to 59.6 hz and is stable.
- 4160V bus voltages are 3840 volts.
- This condition has existed for the past hour.

Which one of the following completes the statement below?

The generator temperatures will (1), AOP-52, Degraded Grid, will require the crew to (2)

| | (1) | (2) |
|---|---|---|
| A. | Rise | Immediately enter AOP-17.0, Rapid Load Reduction. |
| B. | Lower | Immediately enter AOP-17.0, Rapid Load Reduction |
| C. | Rise | Place the unit in Mode 3 in the next six hours using UOP-3.1, Power Operation. |
| D. | Lower | Place the unit in Mode 3 in the next six hours using UOP-3.1, Power Operation. |

Answer: C

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#81** | Tier # | | 1 |
| | Group # | | 1 |
| | K/A # | (BW E04; W E05) Inadequate Heat Transfer—Loss of Secondary Heat Sink / 4<br>WE05EA2.1 - Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): Facility conditions and selection of appropriate procedures during abnormal and emergency operations. | |
| | Importance Rating | | 4.4 |

Proposed Question:

Given the following plant conditions:

- The plant tripped from 100% power.
- The crew has transitioned to FR-H.1, "Response to Loss of Secondary Heat Sink."
- The motor driven AND steam driven EFW pumps tripped and cannot be restored.
- CCP 'A' is running.
- Pressurizer pressure is 2200 psig and increasing.
- Steam Generator wide range levels are:
  - SG 'A': 22%
  - SG 'B': 24%
  - SG 'C': 12%
  - SG 'D': 32%
- Containment pressure is 2 psig and stable.

What action is required next?

A. Immediately initiate bleed and feed.

B. Depressurize SG's and feed with condensate pumps.

C. Try to establish start-up feedwater pump flow to SG's.

D. Check that Pressurizer pressure is less than 2385 psig.

| Proposed Answer: | A | |
|---|---|---|

Explanation (Optional):

A is correct. Per FR-H.1 , Key Notes and Cautions, If wide range level in any 3 SG's is less than 30% (51% for adverse containment) OR pressurizer pressure is greater than or equal to 2385 PSIG due to a loss of secondary heat sink, Steps 10 through 14 should be immediately initiated for bleed and feed. Westinghouse Background Document, FR-H.1, section 2.2, RCS Bleed and Feed Heat Removal states that operator action to establish bleed and feed heat removal can prevent or minimize core uncovery. The document discusses the effectiveness of bleed and feed being dependent on the timeliness of operator action to initiate bleed and feed following indications of the symptoms of loss of all secondary heat sink. It is essential to adhere to this procedural guidance to prevent the adverse effects of core uncovery.

B is incorrect but plausible. The procedure provides extensive guidance for establishing a feed source to the steam generators. The condensate pumps are one of the available feed sources however an attempt would be made to utilize the startup feedwater pump first. Additionally, the criteria for establishing bleed and feed are met.

C is incorrect but plausible. The procedure provides extensive guidance for establishing a feed source to the steam generators. The startup feedwater pumps are one of the available feed sources and would be the next procedurally driven source of feed however the criteria for establishing bleed and feed are met.

D is incorrect but plausible. Guidance for checking Pressurizer pressure is contained in the Key Note/Caution described above for the correct answer, however, given the conditions in the stem of the question the crew should immediately establish bleed and feed based specifically on Steam Generator wide range levels.

| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • FR-H.1, "Response to Loss of Secondary Heat Sink"<br>• Westinghouse Background Document, FR-H.1, Section 2.2, RCS Bleed and Feed Heat Removal, pg. 10 and 68. |
|---|---|

| Proposed references to be provided to applicants during examination: | None |
|---|---|

| Learning Objective: | | | | |
|---|---|---|---|---|
| Question Source: | Bank # | X | Seabrook TEB# 26636 | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive | Memory or Fundamental Knowledge | | | |

| | | | |
|---|---|---|---|
| Level: | Comprehension or Analysis | | |
| 10 CFR Part 55 | 55.41 | | |

| | | |
|---|---|---|
| | 55.43 | |

Comments:

Distractor D was changed from the original distractor which was “D. Do not establish feed flow to any SG. Consult with TSC.” Using the original distractor would cause subject matter overlap with Question #75.

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)?

Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions?

Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs?

Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure?

Yes → RO question

No

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#82** | Tier # | | 1 |
| | Group # | | 2 |
| | K/A # | 005 Inoperable/Stuck Control Rod<br>AA2.03 Ability to determine and interpret the following as they apply to the Inoperable/Stuck Control Rod: Required actions if more than one rod is stuck or inoperable. | |
| | Importance Rating | | 3.5 |

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? (Reference material provided)

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 remainder 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 |
|---|---|

Explanation (Optional):

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.

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.

B 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.

D 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):<br>(Attach if not previously provided)<br>(including version/revision number) | • Tech. Spec. 3.1.3, Movable Control Assemblies Group Height<br>• OS1210.02, Failure of Control Rod or Rod Bank to Move.<br>• OS1210.05, Dropped Rod |

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | Tech. Spec. 3.1.3, Movable Control Assemblies Group Height |

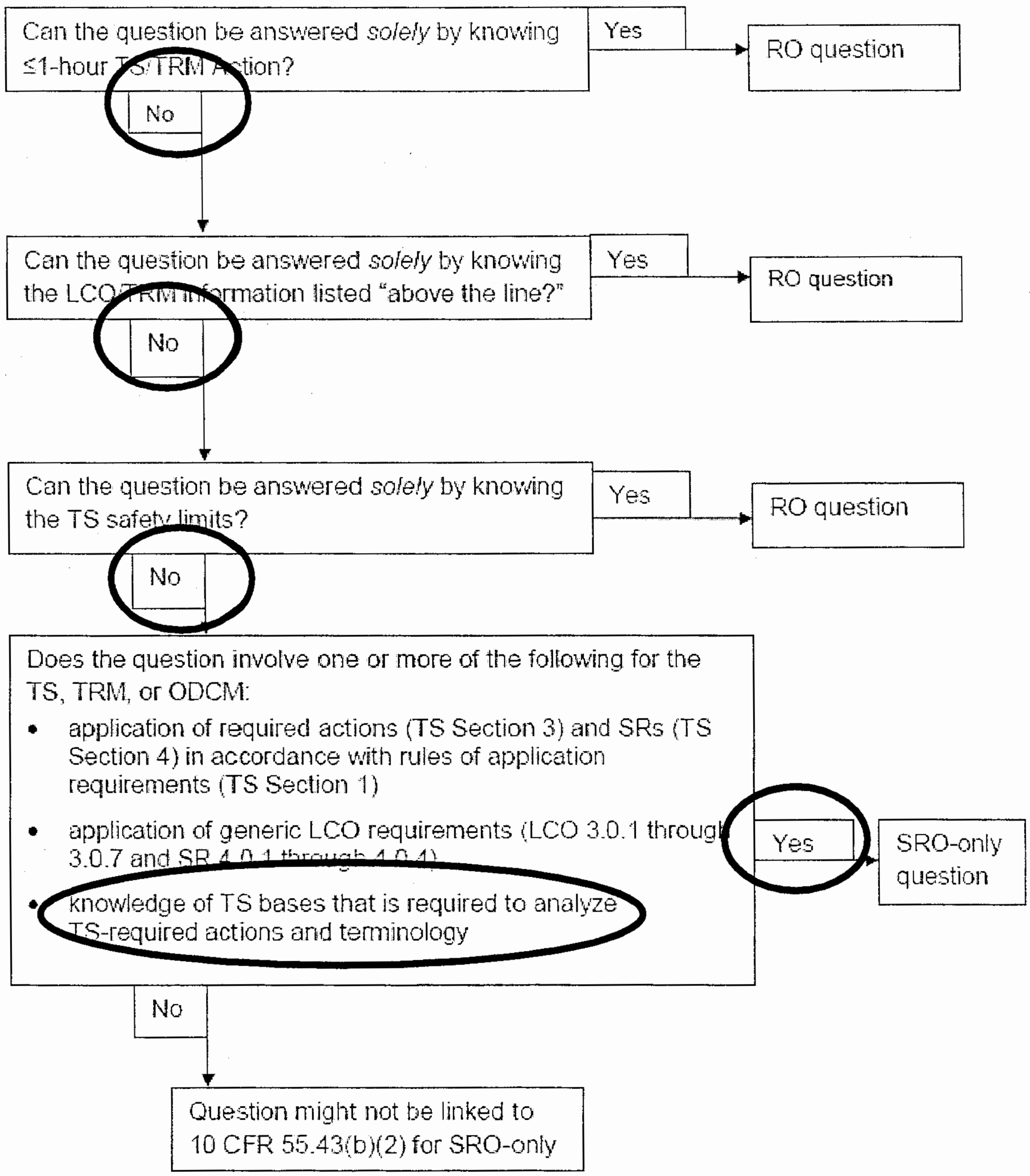
| | | | | |
|---|---|---|---|---|
| Learning Objective: | L8031I24 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2009 NRC Remedial Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | | | |

| | | |
|---|---|---|
| | 55.43 | **43.5** |

Comments:



**Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)**



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#83** | Tier # | | 1 |
| | Group # | | 2 |
| | K/A # | 000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7<br>032AG2.1.30 - Ability to locate and operate components, including local controls. | |
| | Importance Rating | | 4.0 |

Proposed Question:

Given the following conditions:

- The plant tripped 21 minutes ago from full power and is now in MODE 3.
- The crew is implementing ES-0.1, "Reactor Trip Response" and is performing Step 11, "Check If Source Range Detectors Should Be Energized".
- Intermediate Range Nuclear Instrumentation Channel NI-35B is reading $2X10^{-10}$ Amps and stable.
- Intermediate Range Nuclear Instrumentation Channel NI-36B is reading $3X10^{-11}$ Amps and decreasing.
- Power Range Nuclear Instrumentation Channel NI-41B has failed high.

Per ES-0.1, Step 11, what action should be taken and what Tech. Spec. Action applies, if any.

A. Verify that the source range detectors are energized and transfer recorder NR-45 to the source range scale. No Tech. Spec. actions are applicable.

B. Manually energize the source range detectors using the "Source Range High Flux Trip Unblock" pushbuttons. No Tech. Spec. actions are applicable.

C. Manually energize the source range detectors by pulling the control power fuses for Power Range Channel NI-41B. Within 1 hour determine by observation of the associated permissive annunciator window(s) that the associated interlock is in its required state.

D. Manually energize the source range detectors using the "Source Range High Flux Trip Unblock" pushbuttons. Tech Spec action 3.0.3 applies. Within one hour take action to place the plant in at least HOT SHUTDOWN within the next 6 hours.

| Proposed Answer: | B |
|---|---|

Explanation (Optional):

B is correct. The source range channels are designed to automatically energize when both intermediate channels drop below the P-6 reset value of $5X10^{-11}$ amps. Given the conditions in the question stem intermediate range channel 35B is undercompensated and manual action would be necessary to re-energize the source range channels. Per ES-0.1, Step 11 RNO the direction is given to manually re-energize the source range detectors. This action is accomplished using the Source Range High Flux Trip Unblock pushbuttons located on the Main Control Board. Additionally:

- per Tech. Spec. 3.3.1, Reactor Trip System Instrumentation the Source Range detectors meet their surveillance requirements and are OPERABLE
- per Table 3.3-1, "Reactor Trip System Instrumentation", item 5, "Intermediate Range Neutron Flux" is only applicable in MODES 1 and 2.
- per Table 3.3-1, "Reactor Trip System Instrumentation", item 18, "Reactor Trip System Interlocks-Neutron Flux P-6, " is only applicable in MODE 2.

A is incorrect but plausible. The distractor correctly describes that no Tech. Spec. actions are applicable, however given the conditions in the question stem indicate that the intermediate range channel 35B is undercompensated and manual action would be necessary to re-energize the source range channels.

C is incorrect but plausible. It is true that the source range detectors would need to be manually energized because the coincidence was not met for the "P-6" reset value, however the P-6 signal is associated with the intermediate range channels not power range. Additionally, the distractor does describe the action associated with P-6, however per Table 3.3-1, "Reactor Trip System Instrumentation", item 18, "Reactor Trip System Interlocks-Neutron Flux P-6, " is only applicable in MODE 2.

D is incorrect but plausible. The distractor correctly describes the procedural action contained in ES-0.1. Additionally, the student may decide to apply Tech. Spec. 3.0.3. if they misinterpret the applicability of Table 3.3-1, "Reactor Trip System Instrumentation", item 5, "Intermediate Range Neutron Flux", and its associated action (Action 3).

| | | | |
|---|---|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ES-0.1, "Reactor Trip Response"<br>• 1-NHY-509043, NI & Manual Trip Signals<br>• 1-NHY-509044, NI Perm & Blocks<br>• Tech Spec. 3/4.3.1, Reactor Trip System Instrumentation. | | |
| Proposed references to be provided to applicants during examination: | None | | |
| Learning Objective: | L8030I02, L8030I08, L8030I13, L1225I04, | | |
| Question Source: | Bank # | | |

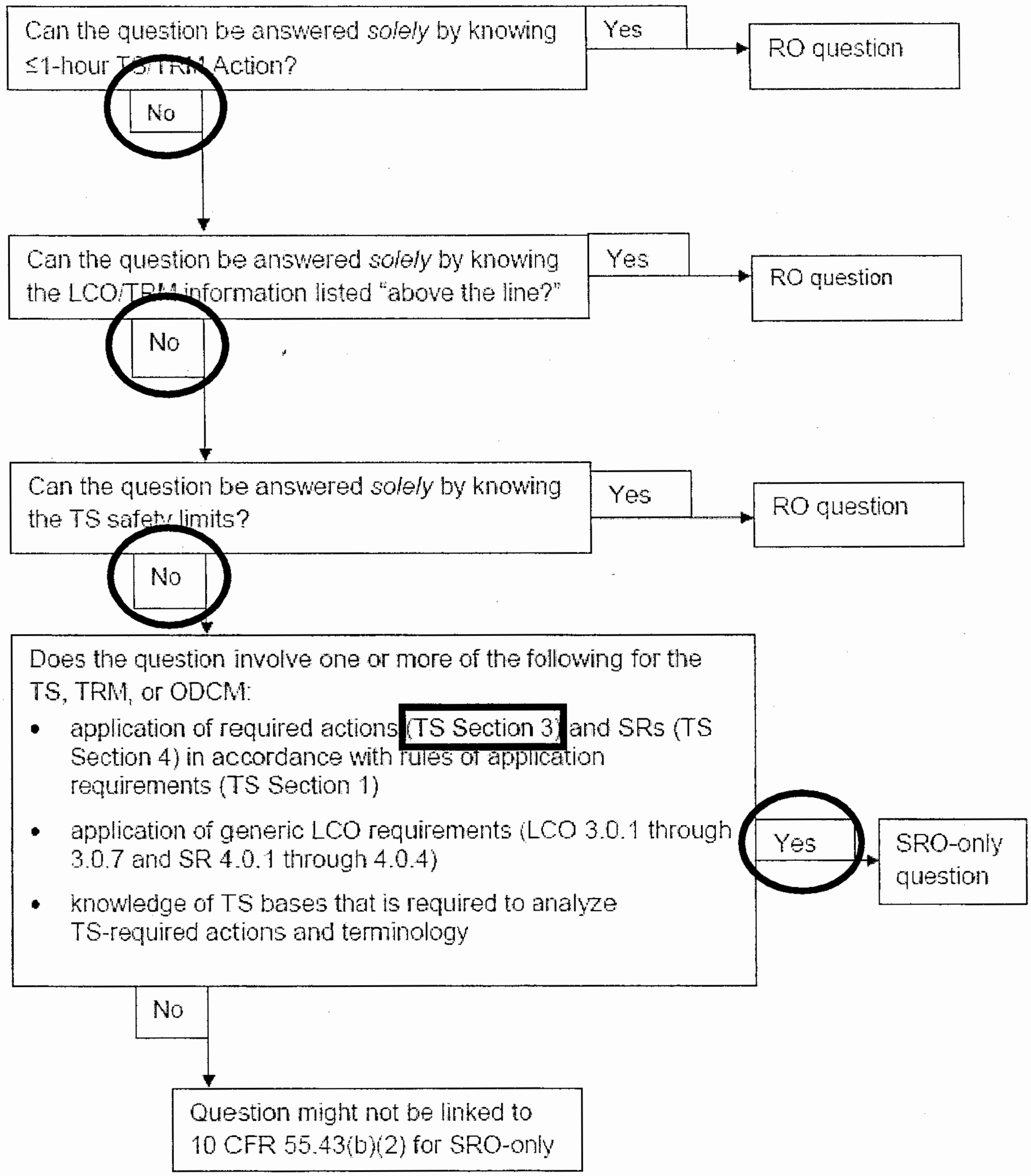
| | | | | |
|---|---|---|---|---|
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 | 55.41 | **41.7** | | |

| | 55.43 | |
|---|---|---|

Comments:



**Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)**



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#84** | Tier # | | 1 |
| | Group # | | 2 |
| | K/A # | 000061 (APE 61) Area Radiation Monitoring System Alarms / 7<br>061AA2.06 – Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Required actions if alarm channel is out of service | |
| | Importance Rating | | 4.1 |

Proposed Question:

Given the following conditions:

- The plant is operating at 100% power with all plant systems in normal alignment.
- While performing the monthly surveillance for RM6562, "Fuel Storage Building Exhaust Monitor" the Radiation Protection Technician notes the following:
  - The Alarm Setpoint is 5.0E+2 Counts Per Minute
  - Background radiation is 2.77E+1 Counts Per Minute

Based on these conditions, what is the MINIMUM required Technical Specification action, if any?

A. No action is required.

B. Adjust the alarm setpoint to within the limit within 4 hours.

C. Declare the radiation monitor alarm inoperable within 72 hours.

D. Restore the required alarm channel to OPERABLE status within 30 days.

| Proposed Answer: | B |
|---|---|

Explanation (Optional):

B is correct. Per Tech. Spec. ¾.3.3, Monitoring Instrumentation, Action a., "With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-6, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable. The Table 3.3-6, Item 4a, "Fuel Storage Building Exhaust Monitor" Alarm/Setpoint column contains the footnote annotation ****. At the bottom of the table the **** notation states "Two times background, or 100 CPM, whichever is greater". Two times background would be 5.54E+1 or 55.4 CPM, thus 100 CPM is the greater of the two vales and is the Tech. Spec. required setpoint limit. The as-found alarm setpoint of 5.0E+2 (500 CPM) is greater than the required limit of 100 CPM, thus the Tech. Spec. action a applies.

A is incorrect but plausible. The student may arrive at this answer if they miscalculate or misinterpret the radiation values as they appear in scientific notation form.

C is incorrect but plausible. It is true that Table 3.3-6 includes a 72 hour action (Action 27), however this action does not apply to the Fuel Storage Building Exhaust Monitor.

D is incorrect but plausible. The required action (Action 25) if the rad monitor is INOPERABLE is a 30 day action, however the most limiting action is the Tech. Spec. main body action a, which allows 4 hours to adjust the setpoint to within tolerance prior to declaring the monitor INOPERABLE.

| | |
|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • Tech. Spec. 3.3.3, Monitoring Instrumentation |

| | |
|---|---|
| Proposed references to be provided to applicants during examination: | Tech. Spec. 3.3.3, Monitoring Instrumentation, including Table 3.3-6. |

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|---|---|---|---|---|
| Learning Objective: | L8059I14 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Beaver Valley 2 2009 NRC Exam | | |

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

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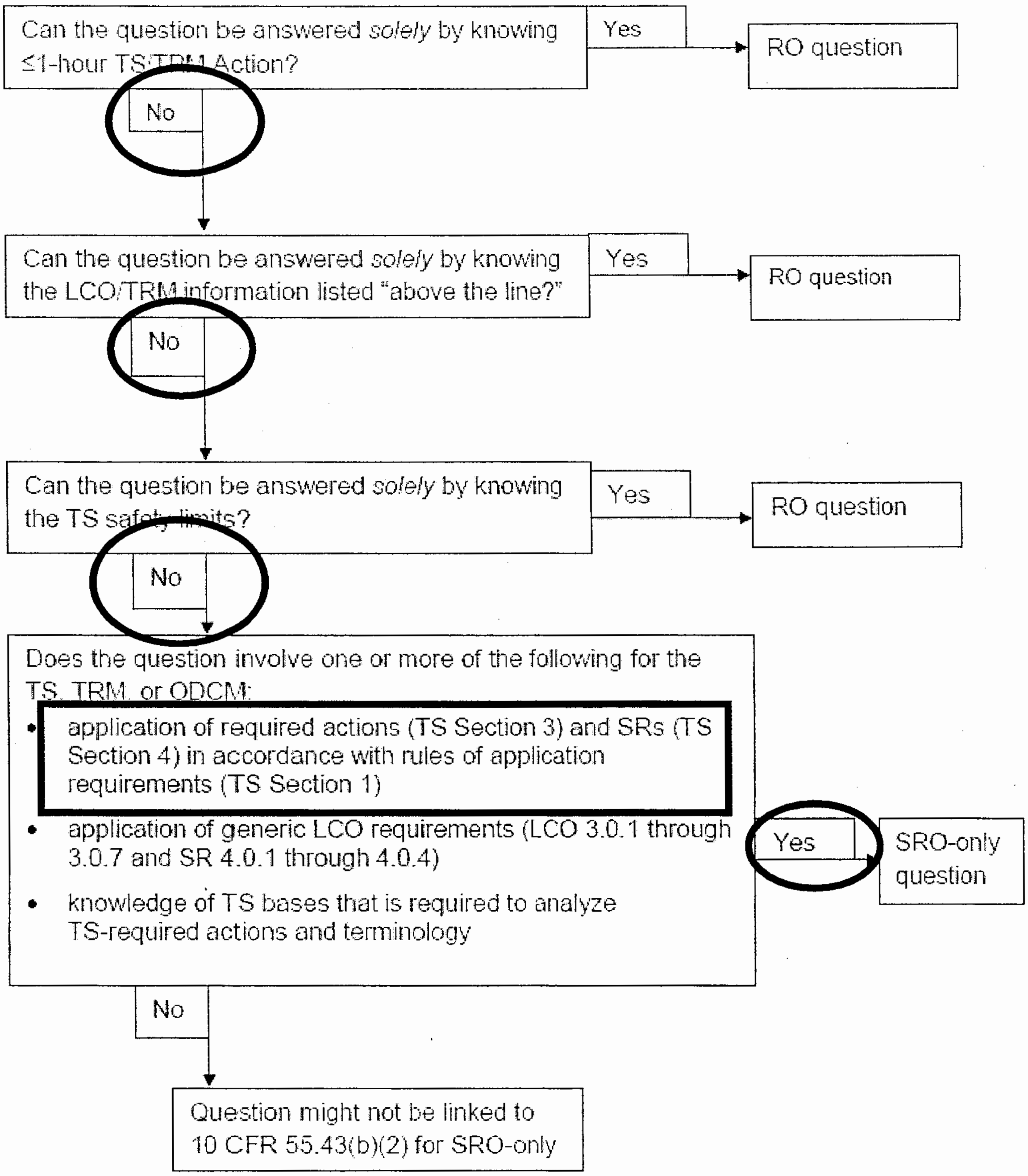
| | 55.43 | **43.5** |

Comments:

Editorial changes were made to the original bank question such the question applied to a Seabrook Station specific radiation monitor. The Tech. Spec. action content, including information for the distractors, remain unchanged from the original bank question.

Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2)
(Technical Specifications)



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#85** | Tier # | | 1 |
| | Group # | | 2 |
| | K/A # | 000069 (APE 69; W E14) Loss of Containment Integrity / 5<br>069AG2.4.20 – Knowledge of operational implications of EOP warnings, cautions and notes. | |
| | Importance Rating | | 4.3 |

| Proposed Question: | |
|---|---|

Given the following conditions:

- The plant was initially at 100% power.
- A reactor trip and automatic Safety Injection occurred.
- The crew entered FR-Z.1, "Response to High Containment Pressure" in response to an ORANGE status on the Containment (Z) status tree.
- The crew is now performing FR-Z.1, Step 5, "Check if Feed Flow should be isolated to any SG".
- All Steam Generators pressures are 300 psig and decreasing in an uncontrolled manner.

What action should the crew take and what is the basis for the action?

| | Action | Basis |
|---|---|---|
| A. | Isolate feed flow to all of the steam generators. | Minimize thermal shock to the steam generators. |
| B. | Maintain at least 25 gallons per minute flow to each steam generator. | Minimize thermal shock to the steam generators. |
| C. | Isolate feed flow to all of the steam generators. | Reduce the cooldown of the Reactor Coolant System. |
| D. | Maintain at least 25 gallons per minute flow to each steam | Reduce the cooldown of the Reactor Coolant |

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| generator. | System. |

| | | |
|---|---|---|
| Proposed Answer: | B | |

Explanation (Optional):

B is correct. The CAUTION statement prior to FR-Z.1, Step 5 states "If all SGs are faulted, at least 25 GPM feed flow should be maintained to each steam generator. Page 15 of the Westinghouse Background Document for FR-Z.1 states "If feed flow to a SG is isolated and the SG is allowed to dry out, subsequent reinitiation of feed flow to the SG could create significant thermal stress conditions on SG components. Maintaining a minimum verifiable feed flow to the SG allows the components to remain in a wet condition, thereby minimizing any thermal shock effects if feed flow is increased.

A is incorrect but plausible. It is true that the basis is to minimize thermal shock. It is plausible for the student to believe that feed flow should be isolated to terminate the thermal effects of the existing feed flow.

C is incorrect but plausible. Since all of the steam generators are depressurizing in an uncontrolled manner it is plausible that isolating feed flow would help reduce RCS cooldown, however that is not the correct basis.

D is incorrect but plausible. It is plausible that the student might recall that 25 GPM flow should be maintained to each SG to keep them "wet" while concurrently reducing RCS cooldown, however that is not the correct basis.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • FR-Z.1, "Response to High Containment Pressure"<br>• Westinghouse Background Document for FR-Z.1, page 15 |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L1212I07, L1212I08 |

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|---|---|---|---|---|
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |

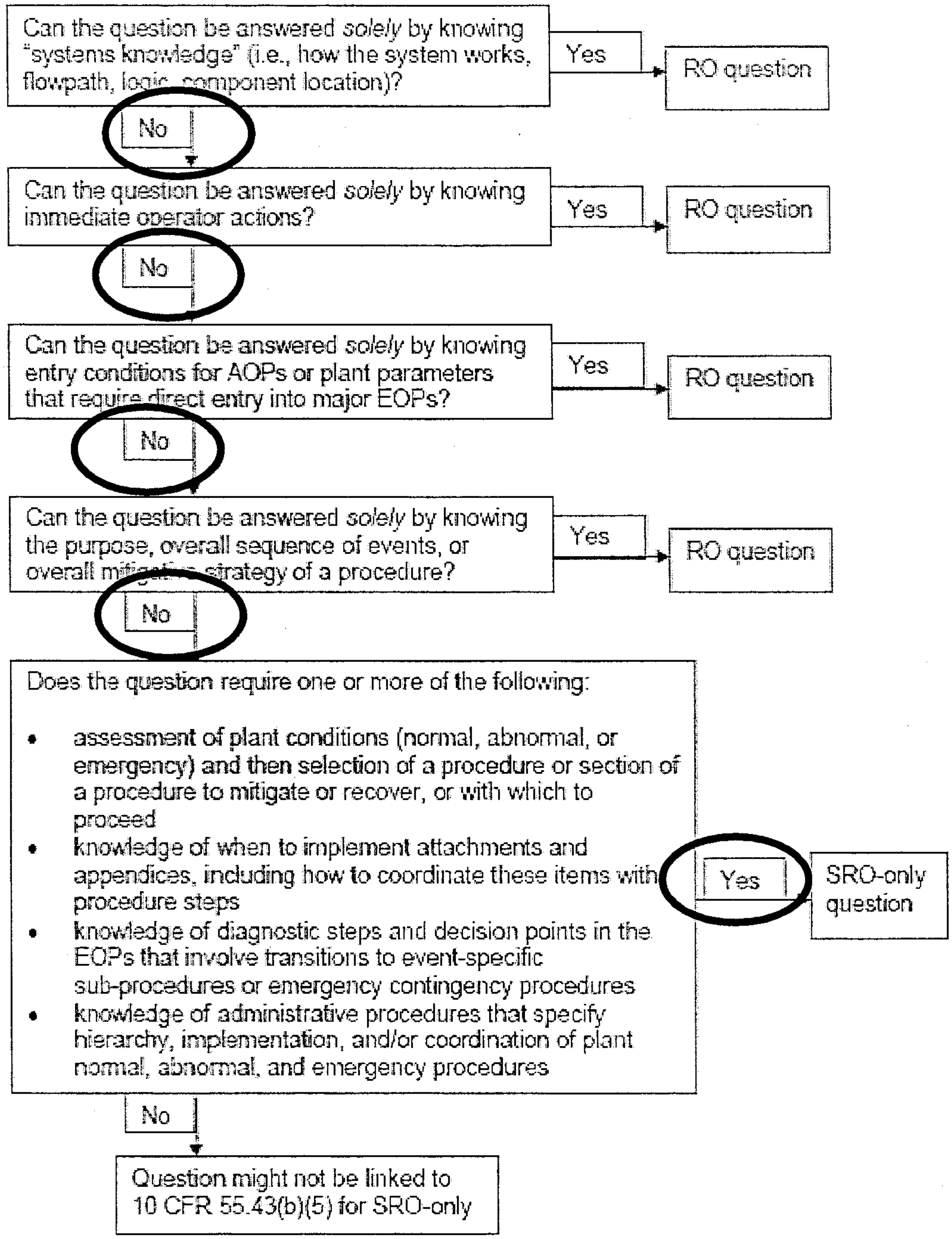
| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X<br>+ | |
| 10 CFR Part 55 | 55.41 | **41.10** | |

| | | |
|---|---|---|
| | 55.43 | **43.5** |

Comments:



Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#86** | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 012 (SF7 RPS) Reactor Protection<br>012A2.05 – 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: Faulty or erratic operation of detectors and function generators | |
| | Importance Rating | | 3.2 |

Proposed Question:

Given the following conditions:

- All actions for removing Power range channel N44 from service have been completed 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 Power Range Channel N41. Coordinate with I&C to troubleshoot channel N41. Place channel N41 associated bistables in the tripped condition within 6 hours.

D. Bypass Power Range 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

Explanation (Optional):

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.<br><br>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.<br><br>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):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1211.04, "Power Range NI Instrument Failure".<br>• Tech. Spec. item ¾. 3.1, Reactor Trip System Instrumentation |
| Proposed references to be provided to applicants during examination: | None |

| | | | | |
|---|---|---|---|---|
| Learning Objective: | L1182I10 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2010 NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | **41.5** | | |

| | | |
|---|---|---|
| | 55.43 | **43.5** |

Comments:



Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)

Can the question be answered *solely* by knowing "systems knowledge" (i.e., how the system works, flowpath, logic, component location)? — Yes → RO question

No

Can the question be answered *solely* by knowing immediate operator actions? — Yes → RO question

No

Can the question be answered *solely* by knowing entry conditions for AOPs or plant parameters that require direct entry into major EOPs? — Yes → RO question

No

Can the question be answered *solely* by knowing the purpose, overall sequence of events, or overall mitigative strategy of a procedure? — Yes → RO question

No

Does the question require one or more of the following:

- assessment of plant conditions (normal, abnormal, or emergency) and then selection of a procedure or section of a procedure to mitigate or recover, or with which to proceed
- knowledge of when to implement attachments and appendices, including how to coordinate these items with procedure steps
- knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures or emergency contingency procedures
- knowledge of administrative procedures that specify hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures

Yes → SRO-only question

No

Question might not be linked to 10 CFR 55.43(b)(5) for SRO-only

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#87** | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | | 013 (SF2 ESFAS) Engineered Safety Features Actuation<br>013A2.03 – Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rapid depressurization |
| | Importance Rating | | 4.7 |

Proposed Question:

Given the following conditions:

- With the plant at 100% power the crew responded to a plant event utilizing the applicable Abnormal Operating Procedure (AOP).
- While responding to the event the following conditions occur:
  - Safety Injection has automatically actuated.
  - The reactor is tripped.
  - Pressurizer level is 15% and lowering.
  - Pressurizer pressure is 1700 psig and lowering.
  - Containment pressure is 2.5 psig and slowly rising.
  - All Steam Generator Pressures are at 1100 psig and slowly lowering.

(1) Which correctly lists Emergency Safety Feature Actuation System (ESFAS) isolation signals that should have occurred, and

(2) What procedure flowpath correctly applies to the given conditions?

| | (1) | (2) |
|---|---|---|
| A. | • Containment Phase “A” Isolation (T Signal)<br>• Main Steam Isolation | OS1201.02, ‘RCS Leak’, E-0, ‘Reactor Trip or Safety Injection’, ES-0.1, ‘Reactor Trip Response’. |

| | | |
|---|---|---|
| B. | • Containment Phase "A" Isolation (T Signal)<br>• Containment Ventilation Isolation (CVI) | OS1201.02, 'RCS Leak', E-0, 'Reactor Trip or Safety Injection', E-1, 'Loss of Reactor or Secondary Coolant'. |
| C. | • Containment Phase "A" Isolation (T Signal)<br>• Main Steam Isolation | OS1227.02, 'Steam Generator Tube Leak', E-0, 'Reactor Trip or Safety Injection', ES-0.1, 'Reactor Trip Response'. |
| D. | • Containment Phase "A" Isolation ("T Signal)<br>• Containment Ventilation Isolation (CVI) | OS1227.02, 'Steam Generator Tube Leak', E-0, 'Reactor Trip or Safety Injection', E-3, 'Steam Generator Tube Rupture'. |

| Proposed Answer: | B |
|---|---|

Explanation (Optional):

B is correct. A Containment Phase "A" Isolation and Containment Ventilation Isolation would occur as they are generated from any SI signal. Additionally, the conditions in the stem of the question indicate that there is a loss of coolant accident (LOCA). The differentiating parameter between a LOCA and a tube leak/rupture is the fact that containment pressure is elevated. Given that the event is a LOCA, and the question stem states that the crew were in the applicable AOP, then OS1201.02, RCS Leak would have entered. Since an automatic Safety Injection occurred, the crew would enter 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).

A is incorrect but plausible. It is true that a Containment Phase "A" Isolation would occur as it is generated from any SI signal, however a Main Steam Isolation signal is not generated by an SI signal and would only occur on low steam line pressure at 585 psig. Given that the event is a LOCA it is true that OS1201.02, RCS Leak would have been entered and given that a safety injection signal occurred it is true that the crew would enter E-0, however, with an SI in progress ES-0.1 would not be entered.

C is incorrect but plausible. It is true that a Containment Phase "A" Isolation and Containment Ventilation Isolation would occur as it is generated from any SI signal, however a Main Steam Isolation signal is not generated by an SI signal and would only occur on low steam line pressure at 585 psig. The procedural flowpath is associated with a steam generator tube leak, however the conditions in the stem of the question indicate that there is a loss of coolant accident (LOCA). The differentiating parameter between a LOCA and a tube leak/rupture is the fact that containment pressure is elevated.

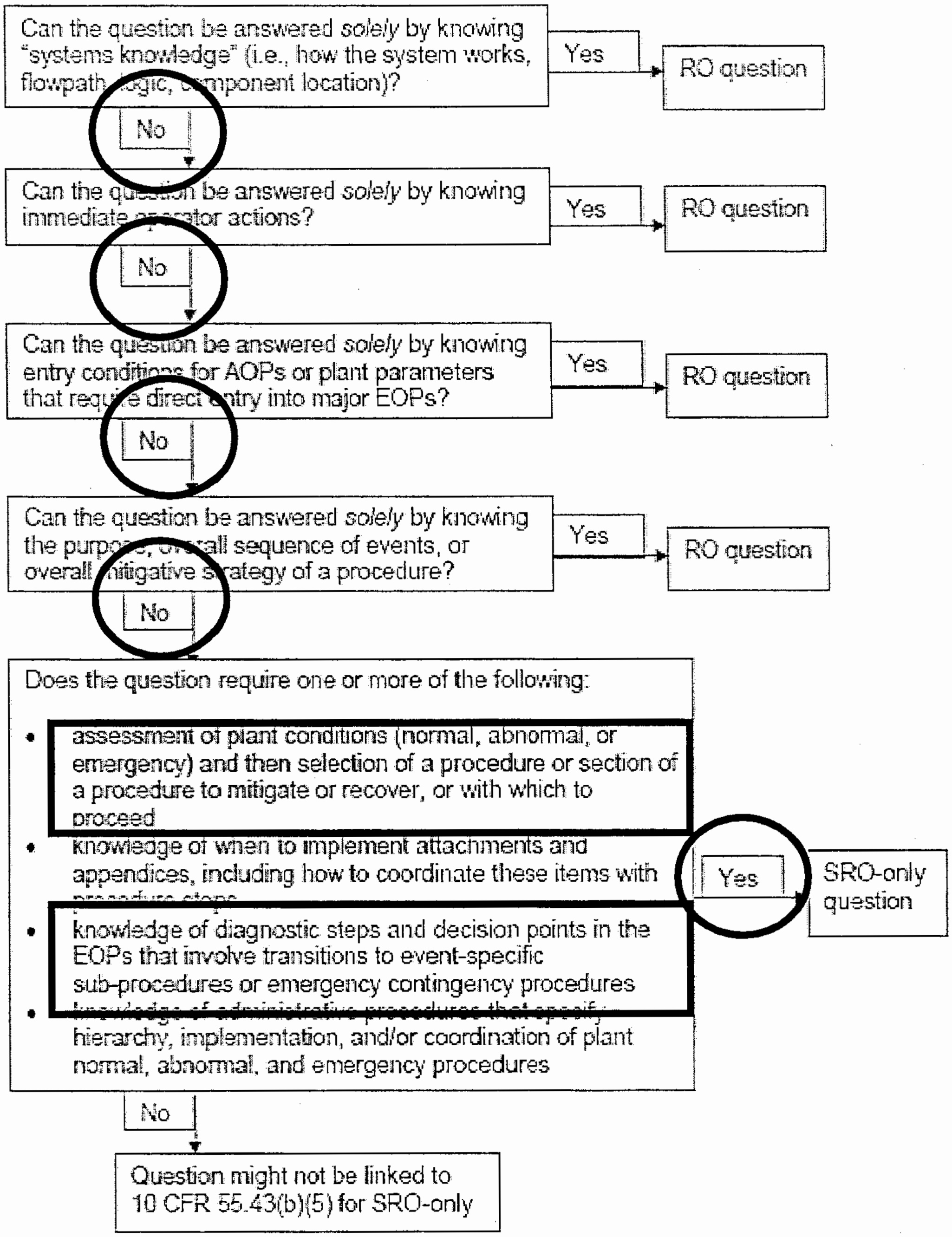
| | | | | |
|---|---|---|---|---|
| D is incorrect but plausible. . It is true that a Containment Phase "A" Isolation and Containment Ventilation Isolation would occur they are generated from any SI signal, however the procedural flowpath is associated with a steam generator tube leak, however the conditions in the stem of the question indicate that there is a loss of coolant accident (LOCA). The differentiating parameter between a LOCA and a tube leak/rupture is the fact that containment pressure is elevated. | | | | |
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1201.02, 'RCS Leak'<br>• E-0, 'Reactor Trip or Safety Injection'<br>• E-1, 'Loss of Reactor or Secondary Coolant'. | | | |
| Proposed references to be provided to applicants during examination: | | | | None |
| Learning Objective: | L1180I05, L1202I08, L1203I01, L1203I02 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 | 55.41 | **41.5** | | |

| | 55.43 | 43.5 |
|---|---|---|

Comments:



Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#88** | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 076 (SF4S SW) Service Water<br>076G2.2.40 - Ability to apply technical specifications for a system. | |
| | Importance Rating | | 4.7 |

Proposed Question:

Given the following conditions:

- 100% power.
- SW-P-41B tripped on overcurrent.
- SW-P-41D did not automatically start, and cannot be manually started by any means.
- OS1216.01, "Degraded Ultimate Heat Sink" is being performed.
- Per step 2, the crew initiated Train "B" Tower Actuation. No SW alignment changes occurred.
- Per step 4, the crew manually aligned SW Train "B" to the Cooling Tower, restoring the Ultimate Heat Sink.
- The Shift Manager is evaluating Technical Specifications.

Which ACTION must be entered, and why? (Reference material provided)

A. ACTION a. One Ocean SW loop is inoperable.

B. ACTION b. One Cooling Tower SW loop is inoperable.

C. ACTION c. Two Cooling Tower SW loops are inoperable.

D. ACTION d. One Ocean SW loop and one Cooling Tower SW loop are inoperable.

Proposed Answer: D

Explanation (Optional):

D is correct. One Ocean SW loop and one Cooling Tower SW loop are inoperable. The Ocean loop is inoperable because no pumps are operable (3.7.4.a). The CT loop is

| |
|---|
| inoperable because none of the components realigned in response to TA actuation (Surveillance requirement 4.7.4.2. b). Action "d" applies.<br><br>A is incorrect but plausible. Action "a" applies to one SW ocean loop only.<br><br>B is incorrect but plausible. Action "b" applies to one CT loop or CT cell.<br><br>C is incorrect but plausible. Action "c" applies to two CT loops or the Tower. |

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • Tech. Spec. 3.7.4, Service Water System/Ultimate Heat Sink |
| Proposed references to be provided to applicants during examination: | Tech. Spec. 3.7.4, Service Water System/Ultimate Heat Sink |

| | | | | |
|---|---|---|---|---|
| Learning Objective: | L8037I14 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |

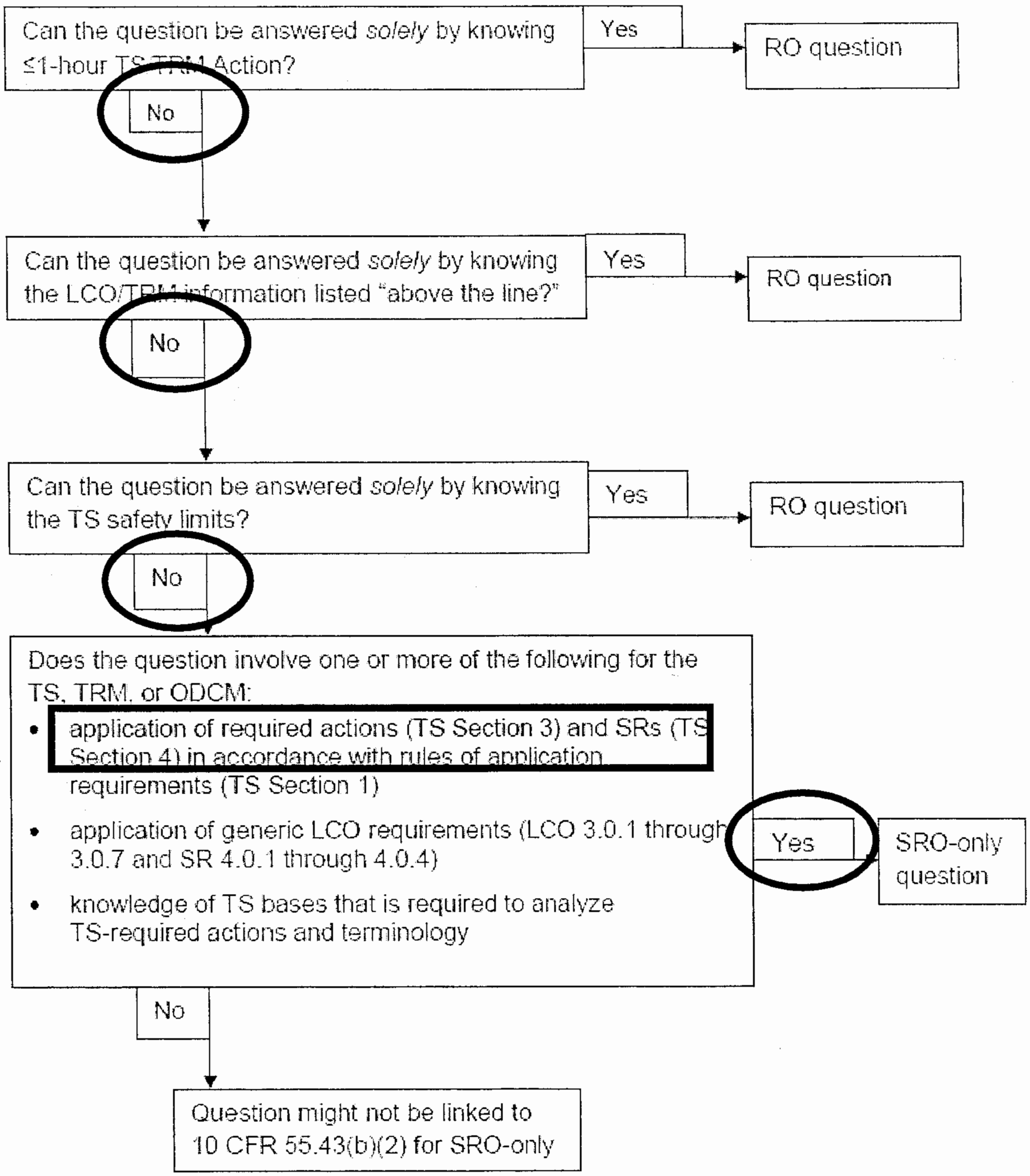
| | | | |
|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | X | |
| 10 CFR Part 55 | 55.41 | **41.10** | |

| | | |
|---|---|---|
| | 55.43 | **43.2, 43.5** |

Comments:



**Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)**



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#89** | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 078 (SF8 IAS) Instrument Air<br>078A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control,or mitigate the consequences of those malfunctions or operations: Air dryer and filter malfunctions | |
| | Importance Rating | | 2.9 |

Proposed Question:

Given the following plant conditions:

- 100% power.
- Containment IA dryer IA-D-2A develops an internal air leak.
- D4985, CONTM INSTRUMENT AIR HDR A PRESS LOW goes into alarm.
- IA-PI-8024, "A" Containment IA header Pressure indicates 95 psig and decreasing.
- ON1242.02, Loss of Containment Instrument Air is entered and corrective actions have been taken.
- "A" Containment IA header pressure indication continues to decrease.

Which of the following describes the correct actions to be taken?

A. Monitor Containment PCCW isolation valve positions.

B. Trip the reactor, Go to E-0, Reactor Trip or Safety Injection. Stop all RCPs.

C. Trip the reactor. Go to E-0, Reactor Trip or Safety Injection. Stop A and D RCPs when immediate actions are complete.

D. Trip the reactor. Go to E-0, Reactor Trip or Safety Injection. Stop B and C RCPs when immediate actions are complete.

Proposed Answer: A

Explanation (Optional):

A is correct. The design of containment instrument air system is such that when cross

connected with the instrument air system by opening IA-V-530 the pressure instrument for the failed loop is isolated by a check valve. There is a note prior to step 1 of ON1242.02 that reminds the operator of this when the procedure is entered. Also containment isolation AOVs inside containment are supplied from both loops of containment instrument air. Even if the 'A' loop depressurized completely valves should remain open. PCCW isolation valves are monitored in the procedure to ensure this occurs and if not a reactor trip is required.

B is incorrect but plausible. An "A" train Phase B isolation signal closes valves in both loops of PCCW supply to containment and would require stopping all RCPs. Loss of instrument air loop 'A' inside containment could be the only supply to 1 valve in each PCCW loop causing loss of cooling flow to all RCPS.

C is incorrect but plausible. Instrument air loop 'A' inside containment could be the only supply to the "A" loop PCCW containment isolation valves IRC. Loss of this supply could result in the loss of cooling to RCPs A and D. There is a common misconception as to which RCPs are cooled by which loop of PCCW.

D is incorrect but plausible. Instrument air loop 'A' inside containment could be the only supply to the "A" loop PCCW containment isolation valves IRC. Loss of this supply could result in the loss of cooling to RCPs B and C. There is a common misconception as to which RCPs are cooled by which loop of PCCW.

| | | | | |
|---|---|---|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • ON1242.02, Loss of Containment Instrument Air<br>• PID-1-IA-B20643, Instrument Air Containment Building | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L1194I06RO | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | **2013 Seabrook NRC Exam: One of previous two previous exams.** | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | | | |
| | 55.43 | **43.5** | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#90** | Tier # | | 2 |
| | Group # | | 1 |
| | K/A # | 103 (SF5 CNT) Containment<br>103G2.1.30 - Ability to locate and operate components, including local controls. | |
| | Importance Rating | | 4.0 |

Proposed Question:

Given the following plant conditions:

- Refueling is in progress.
- The refueling team is reloading the core inside containment.
- The refueling machine interlock override feature must be used to support core packing.

Per OS1015.04, 'Refueling Machine Operation', which of the following requirements must be met before using the refueling machine override feature?

A. Prior to the first use of the bypass feature the Refueling SRO and the Work Control Supervisor must agree that its use is necessary.

B. The Refueling SRO must act as a spotter and continuously observe the fuel assembly.

C. The Refueling SRO must assume the role of Refueling Machine Operator.

D. Each use of the interlock override feature must be distinctly communicated to the Control Room.

Proposed Answer: D

Explanation (Optional):

A is incorrect but plausible. There must be agreement on the necessity of using the bypass feature however agreement must be reached between the Refueling SRO and the Shift Manager vice the Work Control Supervisor.

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. Use of the bypass feature requires enhanced awareness of Refueling Machine activities. At one time Refueling SROs at Seabrook Station did operate the refueling machine. It is conceivable that the Refueling SRO would be required to operate the machine during this condition however the procedure attachment specifically states that the Refueling SRO directly supervises all actions when the interlock feature is

enabled.

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):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1015.04, Refueling Machine Operation | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | | | | |
| Question Source: | Bank # | X | Seabrook 2013 Company Exam | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 | 55.41 | 41.2, 41.10 | | |

| | | |
|---|---|---|
| | 55.43 | 43.6 |

Comments:

F. <u>Procedures and Limitations Involved in Initial Core Loading, Alterations in Core Configuration, Control Rod Programming, and Determination of Various Internal and External Effects on Core Reactivity</u> [10 CFR 55.43(b)(6)]

Some examples of SRO exam items for this topic include the following:

- evaluation of core conditions and emergency classifications based on core conditions

- administrative requirements associated with low-power physics testing processes

- administrative requirements associated with refueling activities, such as approvals required to amend core loading sheets or administrative controls of potential dilution paths and/or activities

- administrative controls associated with the installation of neutron sources

- knowledge of TS bases for reactivity controls

G. <u>Fuel-Handling Facilities and Procedures</u> [10 CFR 55.43(b)(7)]

Some examples of SRO exam items for this topic include the following:

- refuel floor SRO responsibilities
- assessment of fuel-handling equipment SR acceptance criteria
- prerequisites for vessel disassembly and reassembly
- decay heat assessment
- assessment of SRs for the refueling mode
- reporting requirements
- emergency classifications

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#91** | Tier # | | 2 |
| | Group # | | 2 |
| | K/A # | 001 (SF1 CRDS) Control Rod Drive<br>001A2.18 - Ability to (a) predict the impacts of the following malfunction or operations on the CRDS- and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Incorrect rod stepping sequence | |
| | Importance Rating | | 3.8 |

| Proposed Question: | |
|---|---|

Given the following conditions:

- The plant is at 100% power.
- The crew is performing OX1410.02, "Quarterly Rod Operability Surveillance".
- Per RE-20, "RCCA Full Out Positions" the current expected "full out position" is 231 steps.
- When the crew prepares to perform Control Bank "A" testing the control board operator identifies that the Control Bank "A" Group 1 demand counter is at 231 steps and the Group 2 demand counter is at 232 steps.

Which of the following describes the status of Control Bank "A" and what procedural action will correct the condition?

A. Control Bank "A" Groups 1 and 2 are both physically positioned at 231 steps. Per OS1210.06, "Misaligned Control Rod" the crew will position the Rod Bank Selector Switch to Control Bank "A", Insert Group 2 to 231 steps and then position Control Bank "A" to the desired "full out position".

B. The previous rod movement for Control Bank "A" was out of sequence causing Control Bank "A" Group 1 to physically lag one step below Group 2. Per OX1410.02 the crew will position the Rod Bank Selector Switch to Control Bank "A", withdraw Group 1 to 232 steps and then position Control Bank "A" to the desired "full out position".

C. Control Bank "A" Groups 1 and 2 are both physically positioned at 231 steps. Per OS1000.05, "Power Increase" Section 4.3, "Aligning Control Rods After Reactor Startup, the crew will position the Rod Bank Selector Switch to Control Bank "A" and withdraw Group 1 to 232 steps and then position Control Bank "A" to the desired "full

out position".

D. The previous rod movement for Control Bank "A" was out of sequence causing Control Bank "A" Group 2 to physically lead Group 1 by one step. Control Bank "A" Groups 1 and 2 are both physically positioned at 231 steps. Per OS1210.02, "Failure of Control Rod or Rod Bank to Move" the crew will position the Rod Bank Selector Switch to Control Bank "A" and insert Group 2 to 231 steps and then position Control Bank "A" to the desired "full out position".

| Proposed Answer: | C | |
|---|---|---|

Explanation (Optional):

C is correct. The physical withdrawal limit for all control rods is 231 steps, thus the Group 2 counter is incorrectly reading 232 steps. In OX1410.02 the note prior to Step 4.1.10.1 states "If the FOP is 231 steps and/or a control rod group bank demand counter is mismatched above 231 steps then OS1000.05, Power Increase (Aligning Control Rods After Reactor Startup) should be referred to for overstepping guidance to align and insert the control rods to the FOP".

A is incorrect but plausible. It is true that both group are physically located at 231 steps, however OS1210.06, "Misaligned Control Rod" contains guidance for correcting a single misaligned rod and not a whole group.

B is incorrect but plausible. It is true that Control Bank "A" will be will be withdrawn such that the Group 1 step counter aligns with Group 2, however the surveillance procedure should not be used. In fact the surveillance procedure states "The bank selector switch should not be moved without verifying that the group 1 and 2 step demand counters are in agreement. If the group 1 and 2 demand counters are not in agreement prior to moving the bank selector switch, the master cycler will be out of sequence. Having the master cycler out of sequence will result in the wrong group within a bank stepping first."

D is incorrect but plausible. It is true that both banks are physically positioned at 231 steps, however OS1210.02, "Failure of Control Rod or Rod Bank to Move" does not provide the procedural guidance for this situation.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OX1410.02, "Quarterly Rod Operability Surveillance".<br>• OS1000.05, "Power Increase" Section 4.3, "Aligning Control Rods After Reactor Startup |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L1158I03, L8031I03, L8031I08 |
| Question Source: | Bank # |

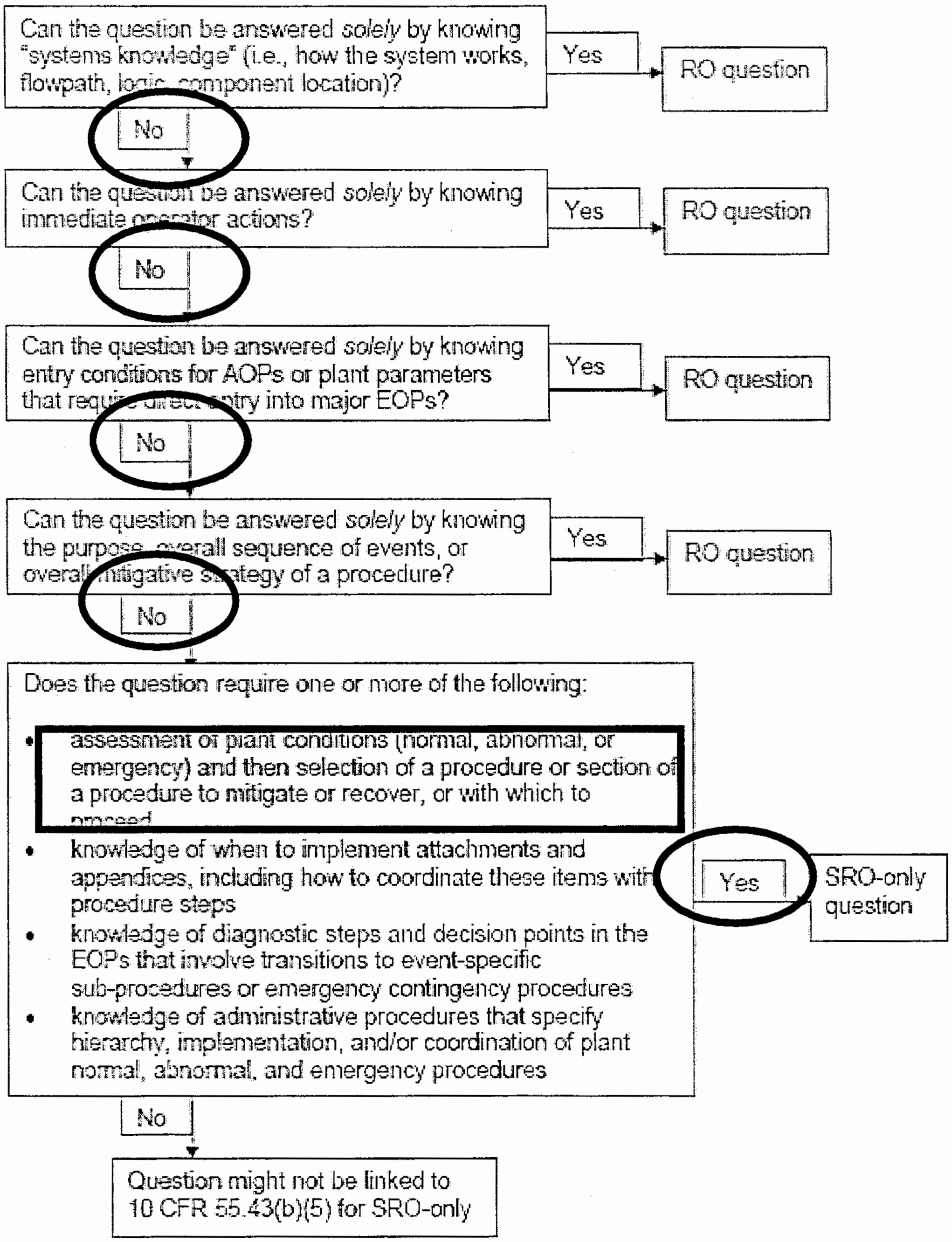
| | | | | |
|---|---|---|---|---|
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 | 55.41 | **41.5** | | |

| | | |
|---|---|---|
| | 55.43 | **43.5** |

Comments:



Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#92** | Tier # | | 2 |
| | Group # | | 2 |
| | K/A # | 015 (SF7 NI) Nuclear Instrumentation<br>015G2.4.9 - Knowledge of low power / shutdown implications in accident (e.g. LOCA or loss of RHR) mitigation strategies. | |
| | Importance Rating | | 4.2 |

Proposed Question:

Given the following conditions:

- The plant was initially at 3% power during a plant startup following a refueling outage.
- An automatic Reactor Trip signal was generated due to a loss of reactor coolant accident (LOCA)
- The reactor did not trip automatically or manually.
- The crew entered FR-S.1, "Response to Nuclear Power Generation".
- The crew was unsuccessful at initiating a boration of the Reactor Coolant System (RCS) per Step 4.
- The crew is now performing Step 15, "Verify Reactor Subcritical and notes the following conditions:
  - o Power Range Nuclear Instrumentation flux level are 3.8% and slowly increasing.
  - o Intermediate range flux rate is +.2 decades per minute.
  - o Gammametrics intermediate range flux level is 4% and slowly increasing.
  - o Gammametrics intermediate range flux rate is +.2 decades per minute.

What procedural actions should the crew take next?

A. Continue efforts to Emergency Borate, exit FR-S.1, and return to procedure and step in effect.

B. Allow the RCS to heat up, Do not exit FR-S.1 or perform any other Functional Restoration Procedure until intermediate range flux rate is zero OR negative.

C. Continue efforts to Emergency Borate. Do not exit FR-S.1 or perform any other Functional Restoration Procedure until Emergency Boration is established per Step 4.

D. Allow the RCS to heat up, perform actions of other Functional Restoration Procedure in effect that do not cooldown or add positive reactivity, and continue efforts to Emergency Borate per Step 4.

| | | |
|---|---|---|
| Proposed Answer: | D | |

Explanation (Optional):

D is correct. Per FR-S.1, Step 15 RNO if nuclear instrumentation indicates a positive startup rate then the RNO action should be performed. The RNO states that if boration is not available then the RCS should be allowed to heat up. Additionally, the direction is given to perform any actions of other FRPs in effect which do not cooldown or otherwise add positive reactivity. The RNO then provides direction to return to Step 4 to continue efforts to establish a boration.

A is incorrect but plausible. It is true that efforts should be made to establish boration, however the criteria is not met to exit FR-S.1 and return to procedure and step in effect.

B is incorrect but plausible. It is true that the direction is to allow the RCS to heat up however the step allows for actions in other FRPs that do not cooldown or add positive reactivity.

C is incorrect but plausible. It is true that efforts should be made to establish boration, however the step allows for actions in other FRPs that do not cooldown or add positive reactivity.

| | |
|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • FR-S.1, "Response to Nuclear Power Generation". |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L1200I07 |

| | | | | |
|---|---|---|---|---|
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive | Memory or Fundamental Knowledge | | | |

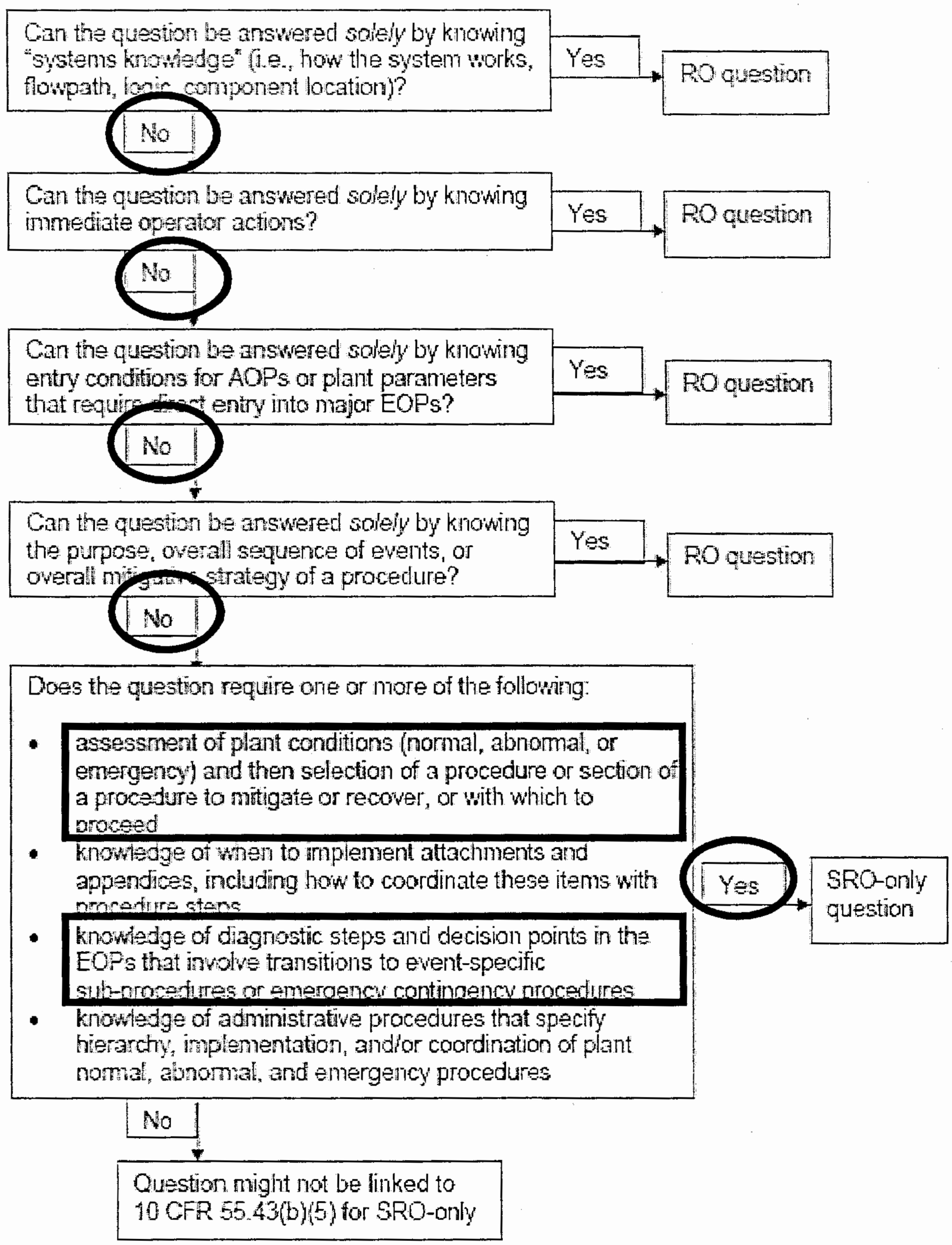
| | | | |
|---|---|---|---|
| Level: | Comprehension or Analysis | X | |
| 10 CFR Part 55 | 55.41 | **41.10** | |

| | 55.43 | **43.5** |
|---|---|---|

Comments:



Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5)
(Assessment and Selection of Procedures)



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#93** | Tier # | | 2 |
| | Group # | | 2 |
| | K/A # | 041 (SF4S SDS) Steam Dump/Turbine Bypass Control<br>041A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the SDS; and (b) based on those predictions or mitigate the consequences of those malfunctions or operations: Steam valve stuck open | |
| | Importance Rating | | 3.9 |

Proposed Question:

Given the following conditions:

- The plant is performing a reactor startup.
- Reactor power is currently stable at 6%.
- The "A" Main Feed Pump is supplying feedwater to the Steam Generators.
- Steam dumps are controlling Tavg in the Steam Pressure mode.
- Subsequently one of the steam dump valves fails to the full open position.
- Reactor Coolant Tavg lowers to 548°F and continues to lower.

Which of the choices completes the following statement?

If RCS Tavg cannot be restored to > 551°F within 15 minutes the reactor trip breakers must be opened within the next 15 minutes because ________________.

A. minimum shutdown margin cannot be assured.

B. calculations determining DNBR are no longer valid.

C. reactor trip instrumentation may not be within its normal range.

D. control rod worth is calculated only over the range of programmed Tavg.

Proposed Answer: C

Explanation (Optional):

C is correct. Per the bases for T.S. 3/4.1.1.4 the specification ensures that the reactor will not be made critical with the RCS temp < 551 °F . This limitation is required to ensure 1) the MTC is within its analyzed temp. range., 2) the trip instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a

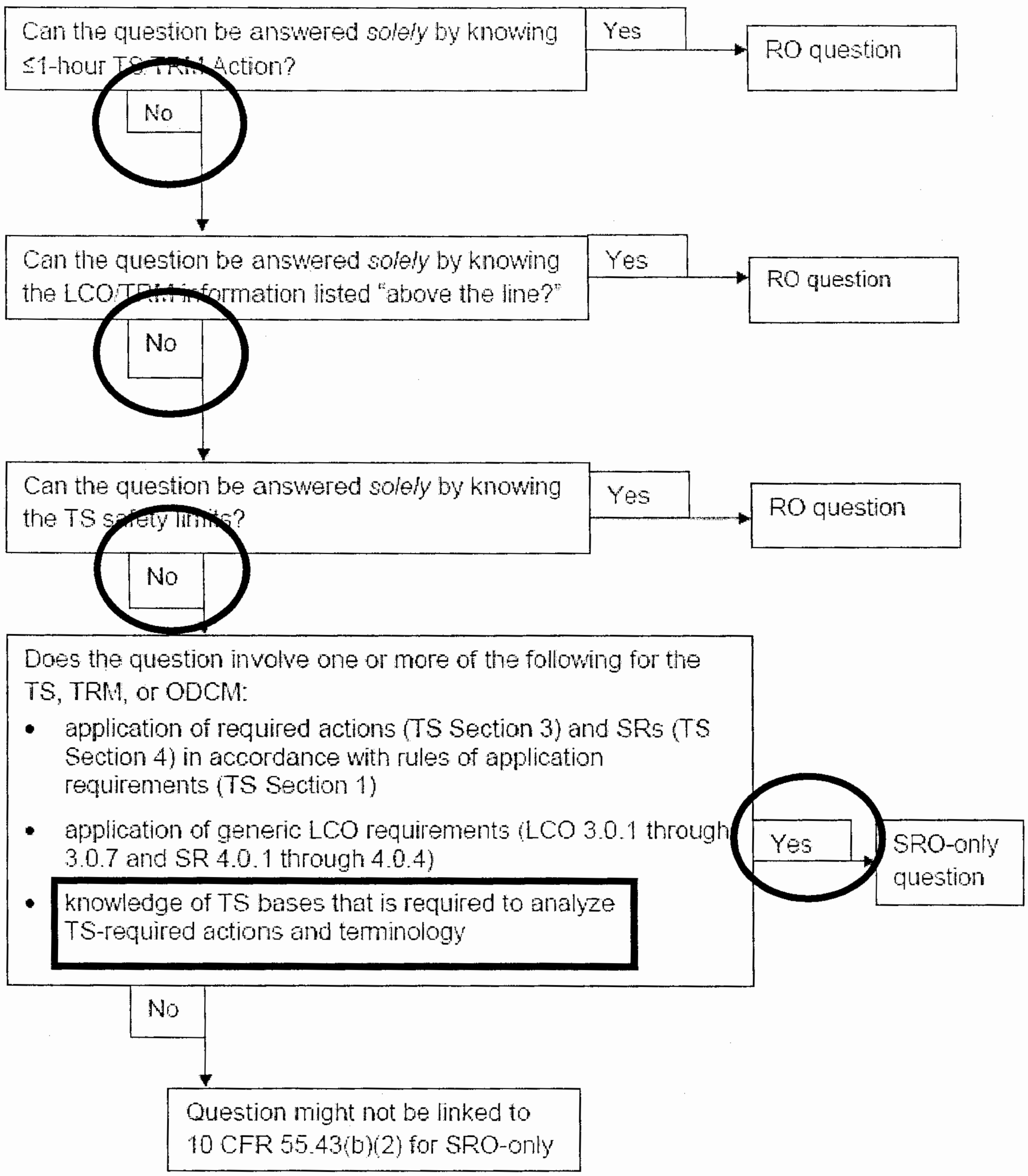
| | | | | |
|---|---|---|---|---|
| steam bubble, and 4) the reactor vessel is above its $RT_{NDT}$ temperature.<br><br>All of the distractors are plausible as each of them describes concepts that could be tied to RCS temperature. | | | | |
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • Tech Spec. 3/4.1.1.4, bases, page B ¾ 1-2. | | | |
| Proposed references to be provided to applicants during examination: | None | | | |
| Learning Objective: | L8010I08 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Salem 2011 NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 | 55.41 | **41.5** | | |

| | | |
|---|---|---|
| | 55.43 | **43.5** |

Comments:



**Figure 2-1 Screening for SRO-Only Linked to 10 CFR 55.43(b)(2) (Technical Specifications)**



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#94** | Tier # | | 3 |
| | Group # | | 1 |
| | K/A # | G2.1.25<br>Ability to interpret reference materials such as graphs, monographs and tables which contain performance data. | |
| | Importance Rating | | 4.2 |

Proposed Question:

Given the following conditions:

- Because the control room became inhabitable the crew is implementing OS1200.02, "Safe Shutdown and Cooldown from the Remote Safe Shutdown Facilities".
- The crew is performing Step 22, "Check CST Inventory" utilizing Attachment C (Reference provided).
- Remote Safe Shutdown Panel instruments indicate that Emergency Feedwater (EFW) flow is 200 gpm per steam generator.
- The NSO reports the following data:
  - The motor driven EFW pump is tagged out and not available.
  - EFW recirculation flow (as read on FW-FI-4279) is 200 gpm.
  - EFW suction pressure ( as read on FW-PI-4208) is 6 psig.
- The Demineralized Water and Water Treatment Systems are unavailable.

What is Condensate Storage Tank (CST) level and what action should be taken?

| | CST Level | Action |
|---|---|---|
| A. | 240,000 gallons | Makeup to the CST as soon as possible from the Fire Protection System. |
| B. | 240,000 gallons | If CST level drops below 212,000 gallons then makeup to the CST from the Cooling Tower Portable Makeup Pump. |
| C. | 285,000 gallons | Makeup to the CST as soon as possible from |

| | | |
|---|---|---|
| | | the Fire Protection System. |
| D. | 285,000 gallons | If CST level drops below 212,000 gallons then makeup to the CST from the Cooling Tower Portable Makeup Pump. |

| Proposed Answer: | C | |
|---|---|---|

Explanation (Optional):

C is correct. With one EFW pump running the CST level calculation is performed using Attachment C, Step 2. Given the information in the stem of the question the interpreted level from the graph "CST Level VS EFW Pump Suction Pressure (1 Pump Operating) is 285,000 gallons.

There is an CAUTION statement prior to procedure step 22 that states "CST makeup should commence as early as possible to avoid low inventory problems late in cooldown". Additionally, with the DM and WT systems not available the remaining CST makeup sources listed in Step 22 are the FP system or the Cooling Tower Portable Makeup Pump.

A is incorrect but plausible. It is true that makeup should commence as soon as possible and that the FP system is an available source as listed in Step 22. Additionally, the student may incorrectly determine that the CST level is 240,000 gallons if they errantly utilize the graph "CST Level VS EFW Pump Suction Pressure (2 Pumps Operating).

B is incorrect but plausible. It is true that the Cooling Tower Portable Makeup Pump is an available source as listed in Step 22, however makeup should commence as soon as possible, not when level drops to 212,000 gallons. The value of 212,000 gallons is plausible as it is the Tech. Spec. value associated with inventory needed for cooldown to RHR.

D is incorrect but plausible. It is true that the CST level is 285,000 gallons. It is also true that the Cooling Tower Portable Makeup Pump is an available source as listed in Step 22, however makeup should commence as soon as possible, not when level drops to 212,000 gallons. The value of 212,000 gallons is plausible as it is the Tech. Spec. value associated with inventory needed for cooldown to RHR.

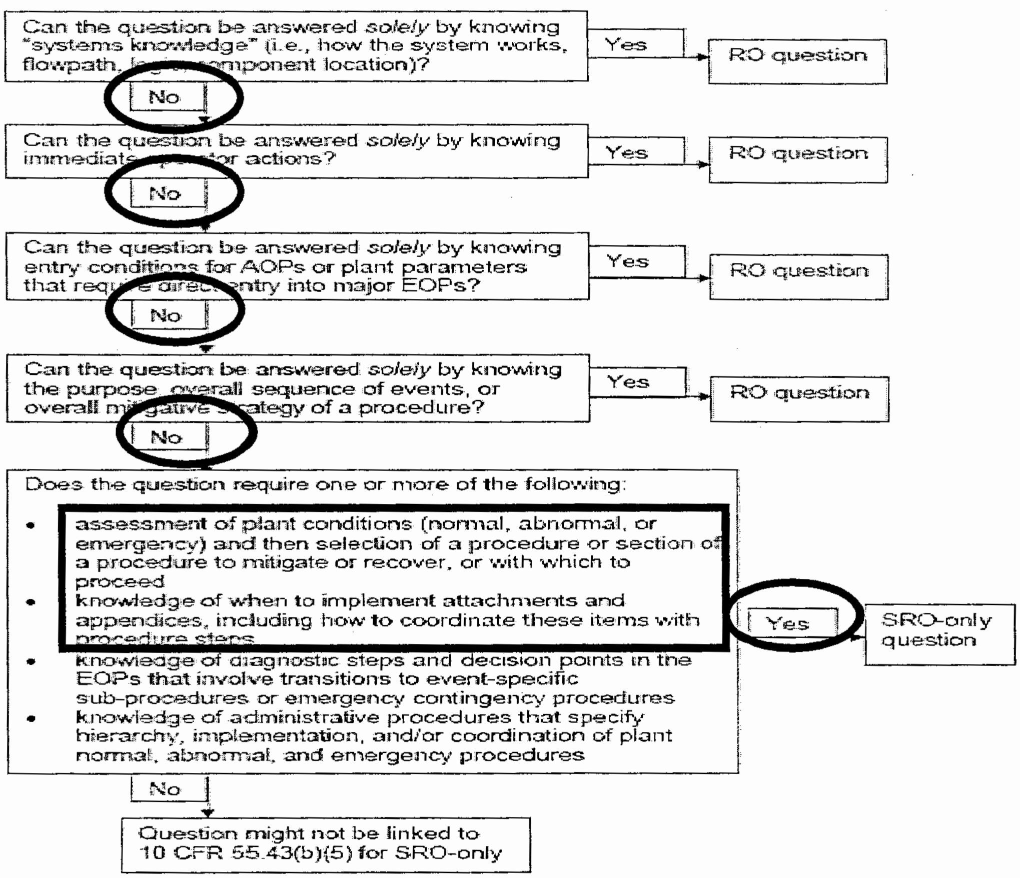
| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • OS1200.02, "Safe Shutdown and Cooldown from the Remote Safe Shutdown Facilities" |
| Proposed references to be provided to applicants during examination: | OS1200.02, "Safe Shutdown and Cooldown from the Remote Safe Shutdown |

| | | | | |
|---|---|---|---|---|
| | | | | Facilities", "Attachment C" only. |
| Learning Objective: | L8210I05 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | **41.10** | | |
| | 55.43 | **43.5** | | |

Comments:

ES-401 8 Attachment 2

**Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)**



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#95** | Tier # | | 3 |
| | Group # | | 1 |
| | K/A # | G2.1.32<br>Ability to explain and apply system limits and precautions. | |
| | Importance Rating | | 4.0 |

| Proposed Question: | |
|---|---|

Which of the following correctly describes the application of a CAUTION statement that precedes 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 |
|---|---|

Explanation (Optional):

C is correct. OP-9.2 states “In general, NOTES and CAUTIONS apply to the step which they precede.

A is incorrect but plausible. While it is correct that a CAUTION that precedes 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 precedes the first operator action step may apply to the entire procedure.

D is incorrect but plausible. OP-9.2, EOP Users Guide states “A NOTE or CAUTION which precedes the first 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.

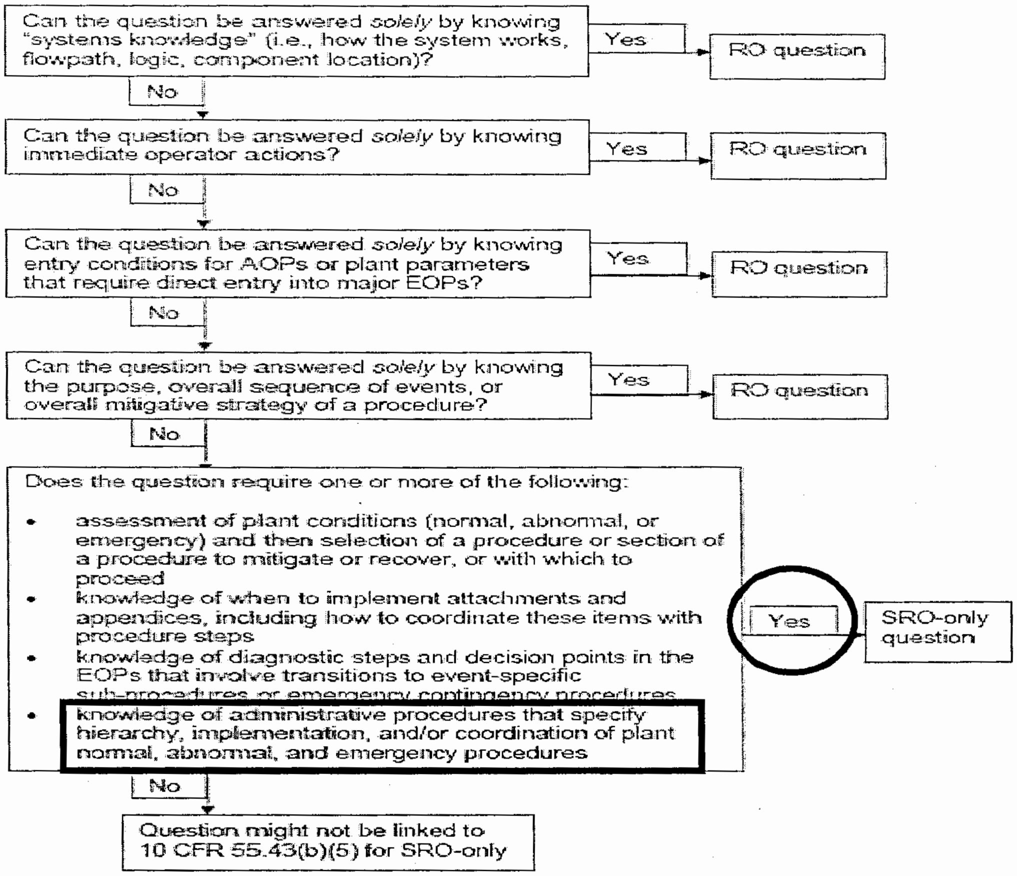
| | | | | | |
|---|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • OP-9.2 EOP Users Guide | | | | |
| Proposed references to be provided to applicants during examination: | | | | None | |
| Learning Objective: | L8069I09 | | | | |
| Question Source: | Bank # | X | Seabrook TEB#34889 | | |
| | Modified Bank# | | | (Note changes or attach Parent) | |
| | New | | | | |
| Question History: | Last NRC Exam | | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | X | |
| | Comprehension or Analysis | | | | |
| 10 CFR Part 55 | 55.41 | 41.10 | | | |

| | | |
|---|---|---|
| | 55.43 | 43.2 |

Comments:

ES-401 8 Attachment 2

Figure 2-2 Screening for SRO-Only Linked to 10 CFR 55.43(b)(5) (Assessment and Selection of Procedures)



| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#96** | Tier # | | 3 |
| | Group # | | 2 |
| | K/A # | G2.2.13<br>Knowledge of tagging and clearance procedures. | |
| | Importance Rating | | 4.3 |

Proposed Question:

When reviewing tagging clearances, what is the MINIMUM temperature and pressure for fluid or gas systems that require verification of double valve isolation?

A. 150°F, 200 psig

B. 200°F, 500 psig

C. 150°F, 500 psig

D. 200°F, 200 psig

Proposed Answer: B

Explanation (Optional):

B is correct. Per MA4.2, “Equipment Tagging and Isolation”, Section 4.3, “Clearance Section Preparation and Review, 4.3.1, “Specific Requirements”, item 6, “When breaching a fluid or gas system that operates with temperatures greater than 200°F or pressures greater than 500 psig, the worker shall have:

- two closed valves in series, and
- an open and tagged telltale vent or drain, between the two valves.

A, C, and D are incorrect but plausible as they all are combinations of the numeric values for the correct answer.

| | |
|---|---|
| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • MA4.2, “Equipment Tagging and Isolation”, Section 4.3, “Clearance Section Preparation and Review, 4.3.1, “Specific Requirements”, item 6 |
| Proposed references to be provided to applicants during examination: | None |
| Learning Objective: | L1305I05 |

| Question Source: | Bank # | X | Seabrook TEB #30012 |
|---|---|---|---|

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<tr><td></td><td>New</td><td></td><td></td><td></td></tr>
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<tr><td colspan="2">Comprehension or Analysis</td><td></td><td></td></tr>
<tr><td rowspan="2">10 CFR Part 55 Content:</td><td>55.41</td><td colspan="3">41.10</td></tr>
<tr><td>55.43</td><td colspan="3"></td></tr>
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</table>

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#97** | Tier # | | 3 |
| | Group # | | 2 |
| | K/A # | G2.2.18<br>Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessment, work prioritizations, etc. | |
| | Importance Rating | | 3.9 |

Proposed Question:

Given the following conditions:

- The plant is in Mode 6.
- Time to boil is 3 hour and 23 minutes.
- DM-V-5, Demin Water Containment Isolation Valve, a manually operated valve located inside containment, is open to support an approved work activity.
- DM-V-4, Demin Water Containment Isolation Valve, a manually operated valve located outside containment, is open and cannot be closed.
- It has been determined that DM-V-5 can be closed within 90 minutes.
- The Spare NSO has been assigned the task of closing the valve and is stationed at the valve.

In accordance with NAWM, "Work Management Manual", Chapter 7, "Outage Risk Management", what is the status of the DM-V-5 PRA closure requirement (reference attached) and why?

A. PRA closure requirements are not applicable.

B. PRA closure requirements are not met because "time to boil" is less than 6 hours.

C. PRA closure requirements are not met because DM-V-5 must be capable of being closed within 1 hour.

D. PRA closure requirements are not met because no manually operated penetration isolations can be open until the core is offloaded.

| Proposed Answer: | C |
|---|---|

Explanation (Optional):

C is correct. Per NAWM, "Work Management Manual", Chapter 7, "Outage Risk

Management", page 7-3.7, PRA closure requirements for open containment penetrations that can only be isolated within containment dictate that the valve:

- shall have a designated trained individual.
- shall have the capability to isolate the penetration within the shorter of TTB or 1 hour.
- shall have an acceptable closure method, and
- shall be constantly manned.

Distractors A, B, and D are all incorrect but plausible as they all pertain to the inventory of requirements as described above.

| | | | | |
|---|---|---|---|---|
| Technical Reference(s): (Attach if not previously provided) (including version/revision number) | • NAWM, "Work Management Manual", Chapter 7, "Outage Risk Management" | | | |
| Proposed references to be provided to applicants during examination: | NAWM, "Work Management Manual", Chapter 7, "Outage Risk Management" | | | |
| Learning Objective: | L1306I08 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | | | |
| | Comprehension or Analysis | X | | |
| 10 CFR Part 55 Content: | 55.41 | **41.10** | | |
| | 55.43 | **43.5** | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#98** | Tier # | | 3 |
| | Group # | | 3 |
| | K/A # | G2.3.12<br>Knowledge of radiological safety principles pertaining to licensed operator duties | |
| | Importance Rating | | 3.7 |

Proposed Question:

The following plant conditions exist:

- Refueling outage is in progress.
- A task is going to be performed in the "A" Steam Generator.
- A Planned Special Exposure (PSE) is going to be used for an individual employee.
- The PSE has been justified as it will reduce the collective doses of all the personnel working on the task.

What reviews and approvals are required when processing the PSE?

A. Health Physics Department Supervisor, Health Physics Department Manager, Operations Manager.

B. Health Physics Department Supervisor, Health Physics Department Manager, Station Director.

C. Shift Manager, Health Physics Department Manager and the Station Director.

D. Health Physics Department Manager and the Station Director. If time permits, obtain verification that the NRC regional office has reviewed the Planned Special Exposure

Proposed Answer: D

Explanation (Optional):

D is correct per RP 5.2, Planned Special Exposures.

A is incorrect but plausible. The planned special exposure does require approval of the HP Dept. Manager. Additional station management approval is needed, however it is from the Station Director vice the Operations Manager.

B is incorrect but plausible. The planned special exposure does require approval of the HP

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| Dept. Manager. Additional station management approval is needed from the Station Director . There is no approval required from the HP Supervisor. HP Supervisor approval is for lower administrative dose approval.<br><br>C is incorrect but plausible. The planned special exposure does require approval of the HP Dept. Manager. Additional station management approval is needed from the Station Director . There is no requirement for approval from the Outage Containment Coordinator. |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • RP5.2, Planned Special Exposures |
| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L1525I13 | | | |
| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |
| Question History: | Last NRC Exam | Seabrook 2010 NRC Exam | | |
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | | |
| | Comprehension or Analysis | | | |
| 10 CFR Part 55 Content: | 55.41 | 41.12 | | |
| | 55.43 | 43.4 | | |
| Comments: | | | | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#99** | Tier # | | 3 |
| | Group # | | 4 |
| | K/A # | G2.4.30<br>Knowledge of events related to system operations/status that must be reported to internal organizations or outside agencies. | |
| | Importance Rating | | 4.1 |

Proposed Question:

Given the following conditions:

- 100% power
- The following maintenance is in progress on "A" train Service Water.
  - SW-P-41A motor replacement.
  - SW-V-2, "SW-P-41-A discharge valve" MOV diagnostics.

Which of the following situations associated with this activity would require immediate state and/or federal notification?

A. 10 gallon oil spill to the SW fore-bay.

B. Maintenance tech injury requiring immediate transport and hospitalization.

C. Operating permit holder connects to MOV diagnostic equipment and closes SW-V-22, "SW-P-41-C discharge valve".

D. 10 gallon oil spill to asphalt outside the SWPH door that is cleaned up immediately and disposed of properly.

| Proposed Answer: | A |
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Explanation (Optional):

A is correct. ON1244.01, Spill Response Attachment 'B' Requires NHDES, US Coast Guard and National Response Center notification of any petroleum spill with a potential to discharge to the ocean. This requirement is contained in LI-AA-102-1001, "Regulatory Reporting", Attachment 4, "Seabrook Site Reports" with the reporting timeline being "ASAP"

B is incorrect but plausible. An injury requiring transport to the hospital is an OSHA

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| reportable event. Per LI-AA-102-1001, "Regulatory Reporting", Attachment 3, "Other Event Related Reports", an incident involving hospitalization of 3 or more employees is required to be reported to OSHA within 8 hours.<br><br>C is incorrect but plausible. Inadvertent closure of SW-V-22 on the running SW pump will case a Tower Actuation signal. Per LI-AA-102-1001, "Regulatory Reporting", Attachment 1, "Reportable Events", this would require an 8 hour report to the NRC as a System Actuation.<br><br>D is incorrect but plausible. ON1244.01, Spill Response Attachment 'B' a spill of < 25 gallons of oil to land where it will ultimately seep into ground water unless the discharge is cleaned up immediately and disposed of properly. |

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number) | • LI-AA-102-1001, "Regulatory Reporting"<br>• ON1244.01, "Spill Response |

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| Proposed references to be provided to applicants during examination: | None |

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| Learning Objective: | L1191I04 |

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| Question Source: | Bank # | X | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | | | |

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|---|---|---|
| Question History: | Last NRC Exam | **2013 Seabrook NRC Exam: One of previous two previous exams.** |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | | |
| | Comprehension or Analysis | | |

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| 10 CFR Part 55 Content: | 55.41 | **41.10** |
| | 55.43 | **43.5** |

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| Comments:<br><br>SRO level justification: Learning objective L1191I04, "Describe the criteria for regulatory notification per Step 4 of ON1244.01, "Spill Response" is designated as an <u>SRO ONLY</u> objective. |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| **Question#100** | Tier # | | 3 |
| | Group # | | 4 |
| | K/A # | G2.4.46<br>Ability to verify that the alarms are consistent with the plant conditions. | |
| | Importance Rating | | 4.2 |

Proposed Question:

Given the following conditions:

- Computer point C0006, "Thermal Power 4 Min Avg" is 3648.2 MWt
- Computer point C0008, "Thermal Power 1 Hr Avg" is 3648.1 MWt

What is the status of plant power level and what action should be taken?

A. The facilities operating license limit has been exceeded due to the average power level for 1 hour exceeding 100% of rated power. Promptly restore the power to no greater than the facility 100% power operating limit.

B. The facilities operating license limit has been exceeded due to the the average power level for 4 minutes exceeding 100% of rated power. Promptly restore the power to no greater than the facility 100% power operating limit.

C. The facilities operating license limit has been exceeded due to the average power level for 4 minutes and 1 hour exceeding 100% of rated power. Promptly restore the power to no greater than the facility 100% power operating limit.

D. The facilities operating license limit has not necessarily been exceeded. In order to prevent exceeding the thermal power license limit the 1 hour average should be reduced below and maintained below the facility 100% power operating limit.

| Proposed Answer: | D |
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Explanation (Optional):

D is correct. The facilities 100% rated thermal power limit is 3648 MWt as defined in Seabrook Station, Unit No.1, Facility Operating License, 2.C (1), "Maximum Power Level.
Per OS1000.10, "Operation At Power"

- the average power level over any 8 hour shift must not exceed 100% (3648 MWt).
- In order to prevent exceeding the thermal power license limit the Operations Manager's expectation is that the one hour average will be maintained below 3648 MWt

A is incorrect but plausible. It is true that power should be restored to below the 100% power limit.

It is plausible that the license limit has been exceeded as the stem of the question presents power values that are in excess of 3648 MWt, however the license limit exceedance is defined by the average power level over any 8 hour shift vice the 1 hour average.

B is incorrect but plausible. It is true that power should be restored to below the 100% power limit. It is plausible that the license limit has been exceeded as the stem of the question presents power values that are in excess of 3648 MWt, however the license limit exceedance is defined by the average power level over any 8 hour shift vice the 4 minute average.

C is incorrect but plausible. It is true that power should be restored to below the 100% power limit. It is plausible that the license limit has been exceeded as the stem of the question presents power values that are in excess of 3648 MWt, however the license limit exceedance is defined by the average power level over any 8 hour shift vice the combination of the 1 hour and 4 minute average.

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| Technical Reference(s):<br>(Attach if not previously provided)<br>(including version/revision number-r) | • Seabrook Station, Unit No.1, Facility Operating License, 2.C (1), "Maximum Power Level.<br>• OS1000.10, "Operation At Power" |

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| Proposed references to be provided to applicants during examination: | None |

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|---|---|---|---|---|
| Learning Objective: | L8010I03 | | | |
| Question Source: | Bank # | | | |
| | Modified Bank# | | | (Note changes or attach Parent) |
| | New | X | | |
| Question History: | Last NRC Exam | | | |

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|---|---|---|---|
| Question Cognitive Level: | Memory or Fundamental Knowledge | X | |
| | Comprehension or Analysis | | |
| 10 CFR Part 55 Content: | 55.41 | **41.10** | |
| | 55.43 | **43.5** | |

Comments:

SRO justification: The content of this question pertains to 10 CFR Part 55.43(b)(1), Conditions and limitations in the facility license.